APPENDIX F
EVALUATION OF HUMAN HEALTH EFFECTS FROM FACILITY ACCIDENTS

F.1 INTRODUCTION

This appendix presents the methodology and assumptions used for estimating potential impacts and risks associated with both radiological and toxic chemical releases, due to postulated accidents, at the facilities being considered for the treatment and management of sodium-bonded spent nuclear fuel. Analysis of radiological impacts is presented in Section F.2. This is followed by a summary of the risk results for the various alternatives. Chemical risk methodologies and results are presented in Section F.3. Information regarding the impacts of normal operations, along with background information on the health impacts from exposure to ionizing radiation, is provided in Appendix E.

F.2 IMPACTS OF RADIOLOGICAL ACCIDENTS ON HUMAN HEALTH

This section addresses the radiological impacts associated with accidents at management facilities. Potential accident scenarios have been identified for both the Argonne National Laboratory-West (ANL-W) and Savannah River Site (SRS) facilities proposed for the treatment and management of sodium-bonded spent nuclear fuel.

F.2.1 Overview of Methodology and Basic Assumptions

For the radiological evaluation, the GENII computer program (PNL 1988) was used to calculate radiation doses to the general population and selected individuals. Appendix E provides the detailed description of this code; therefore, only the GENII data specific to the accident analysis is presented in this appendix.

The impacts of radiation exposure were evaluated for the following population segments for each accident scenario:

- *Noninvolved Worker*—An individual located 100 meters (330 feet) from the radioactive material release point.[1] The dose to the noninvolved worker was calculated for the 50th percentile meteorology only, as specified in the U.S. Department of Energy (DOE) Hazard Categorization and Accident Analysis Techniques Standard (DOE 1992). Noninvolved workers would be exposed unprotected to the plume for a limited time (a maximum of 5 minutes), receiving exposure via inhalation, air immersion, and ground surface pathways only.

- *Maximally Exposed Offsite Individual*—An individual member of the public living at the management site boundary and receiving the maximum exposure. This individual is located directly downwind of the accident and would be exposed to radioactivity via inhalation, ingestion, air immersion, and ground surface pathways. The individual would be exposed to the plume for the entire release duration.

- *Population*—The general public living within an 80-kilometer (50-mile) radius of the facility, residing directly downwind of the accident, and receiving the maximum exposure via inhalation, ingestion, air immersion, and ground surface pathways.

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[1] *For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose could occur is frequently greater than 100 meters (330 feet) for an elevated release.*

The doses to the maximally exposed offsite individual and the general public were calculated for the 50th and 95th percentile meteorological conditions. Meteorology specific to ANL-W and SRS was used in the evaluation. Site-specific meteorological data was obtained in the form of a joint frequency distribution in terms of percentage of time that the wind blows in specific directions for the given midpoint (or average) wind speed and atmospheric stability. Accident consequences were calculated for both 50th and 95th percentile meteorological conditions. The 50th percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95th percentile condition represents relatively low-probability meteorological conditions that produce higher calculated exposures, and is defined as that condition not exceeded more than 5 percent of the time. GENII determines 50th and 95th percentile meteorological conditions using site-specific joint frequency distribution weather data.

The following conditions were used in the calculations:

- Meteorological Data
  - Site-specific joint frequency distribution weather data were used to define 50th and 95th percentile meteorological conditions for each processing technology at management sites.
  - Any release through a stack was assumed to occur at an elevated level consistent with the site's effluent emission stack height. The effects of plume rise were not credited in the analysis.
  - Mixing layer height is 1,000 meters (3,280 feet). Airborne materials freely diffuse in the atmosphere near the ground level in what is known as the "mixing depth." A stable layer exists above the mixing depth and restricts vertical diffusion above 1,000 meters (3,280 feet).
  - Wet deposition is zero (it was assumed that no rain occurs to accelerate deposition and reduce the size of the area affected by the release).
  - Dry deposition of the cloud was modeled. During movement of the radioactive plume, a fraction of the radioactive material in the plume is deposited on the ground due to gravitational forces. The quantity of deposited radioactive material is proportional to the particle size and deposition velocities (in meters per second). The deposited material contributes to the exposure from ground surface radiation and ingestion.
- Inhalation Data
  - Breathing rate is 330 cubic centimeters per second (0.7 cubic feet per minute) for the worker and the general public at the site boundary and beyond (maximally exposed offsite individual and population) during the passage of the plume; it is 270 cubic centimeters per second (0.57 cubic feet per minute) for the general public during the other times.
  - Exposure during passage of the entire plume was assessed for the maximally exposed offsite individual and the population. Exposure to the noninvolved worker is to a portion of the plume (i.e., the noninvolved worker is exposed to the plume for a limited time) because the worker is assumed to take emergency action.
  - Inhalation exposure factors are based on the International Commission on Radiological Protection, Publication 30 (ICRP 1982).

Exposure time assumptions for maximally exposed offsite individuals, workers, and the general public are provided in **Table F–1** below. Since all accident releases would be to the air (either gaseous or suspended particulates), drinking water, aquatic food ingestion, and any other pathways that may involve liquid exposure were not examined. Additional information, common to the analysis of normal operation and accident impacts, is presented in Appendix E.

Table F–1 GENII Plume and Soil Contamination Exposure Parameters (Postulated Accidents)

Maximally Exposed Offsite Individual			*General Population*		
Inhalation and External Exposure			*Inhalation and External Exposure*		
Exposure Time (hours)	*Breathing Rate (cubic centimeters per second)*	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cubic centimeters per second)*	*Soil Contamination (hours)*
100 percent of release time	330	6,136	100 percent of release time	330	6,136

Source: PNL 1988.

Radiological impacts to noninvolved workers from postulated accident scenarios were evaluated at onsite locations where a given incident would cause the highest dose. The noninvolved worker was assumed to have an inhalation exposure time of 5 minutes and an external exposure time to soil contamination of 20 minutes. For a ground-level release accident, a noninvolved worker was assumed to be 100 meters (330 feet) from a given release point; for an elevated release, the worker was situated between 200 and 500 meters (660 and 1,640 feet), depending on the given site's atmospheric dispersion characteristics. All doses to noninvolved workers include a component associated with the intake of radioactivity into the body and another component resulting from external exposure to direct radiation.

The radiation doses to individuals and the public resulting from exposure to radioactive releases were calculated using the following potential pathways:

- *Air immersion*—External direct exposure from immersion in the airborne radioactive material
- *Ground surface*—External direct exposure from radioactive material deposited on the ground
- *Inhalation*—Internal exposure from inhalation of radioactive aerosols and suspended particles
- *Ingestion*—Internal exposure from ingestion of contaminated terrestrial food or animal products

The radiation doses were estimated by the GENII computer program, which uses the dose models recommended by the International Commission on Radiological Protection in Publications 26 and 30 (ICRP 1977, ICRP 1982). Committed dose equivalents[2] are calculated individually for organs such as the gonads, breast, red bone marrow, lungs, thyroid, and bone surface; calculations are combined for the liver, upper large intestine, lower large intestine, small intestine, and stomach. Weighting factors are used for various body organs to calculate weighted or committed effective dose equivalents from radiation inside the body due to inhalation or ingestion. The committed effective dose equivalent value is the sum of the committed dose equivalent to a specific organ weighted by the relative risk to that organ compared to an equivalent whole-body exposure. The deep-dose equivalent for the external exposure pathways (immersion in the radioactive material and exposure to the ground contamination) and the 50-year committed effective dose equivalent for the internal exposure pathways were calculated. The sum of the deep-dose equivalent for external pathways and the committed effective dose equivalent for internal pathways is called the "total effective dose equivalent," or simply, the "total dose" in this environmental impact statement (EIS).

[2] *The definitions of committed dose equivalents, committed effective dose equivalents, and total effective dose equivalents are consistent with those given in 10 CFR Part 835, "Occupational Radiation Protection; Final Rule."*

The exposure from ingestion of contaminated terrestrial food or animal products is calculated on a yearly basis. It is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose should exceed that of the protective action guidelines in a radiological accident event (EPA 1991). No reduction of exposure because of protective actions or evacuation of the public was accounted for in this analysis, however. This conservative approach may result in overestimating health effects within an exposed population, but allows for consistent comparison between alternatives.

F.2.2 Selection of Facility Accidents for Detailed Evaluations

The alternatives for the treatment of sodium-bonded spent nuclear fuel assume the use of facilities currently in operation, although modifications to SRS Building 105-L would be necessary before it could be used for the melt and dilute alternative. The selection of accident scenarios was based on those evaluated in the safety analysis reports for the facilities.

Postulated facility accident scenarios were developed based on the review of the analyzed accidents in previous safety analysis, risk assessment, and environmental assessment documents at ANL-W and SRS, where the sodium-bonded spent nuclear fuel may be handled or processed. These documents include the following:

- *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995a)

- *Environmental Assessment Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at ANL-West* (DOE 1996a)

- *Fuel Cycle Facility Final Safety Analysis Report* (ANL 1998a)

- *Safety Analysis Report for the Hot Fuel Examination Facility* (ANL 1998b)

- *Accident Assessments for Idaho National Engineering Laboratory Facilities* (Slaughterbeck et al. 1995)

- *Safety Analysis-200 Area, Savannah River Site F-Canyon Operation, F-Canyon SAR Addendum* (WSRC 1994)

- *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000)

Based on this review of analyzed accident scenarios at ANL-W and SRS facilities that deal with sodium-bonded spent nuclear fuel, a spectrum of potential accidents was identified. This process started with systematically identifying initiating events, subsequent accident progressions, and onsite or offsite releases. Then, based on accident initiators, selected accidents were grouped into the following three categories:

- Natural phenomena (e.g., earthquake, tornado)
- External events (e.g., aircraft crash)
- Process-related events (e.g., explosion, nuclear criticality, fire, spills)

The potential process-related events were further subdivided based on the impact the accident would have on the accident release factors. High-energy events would be expected to damage some of the confinement barriers provided in the facility design and would result in release factors that approach unity. Medium-energy events could reduce the effectiveness of the barriers, but would not be expected to defeat them, while low-energy events would have almost no impact on the ability of the confinement barriers to perform their function.

A review of the accident scenarios indicated that only severe accident conditions (e.g., accidents involving confinement failure) could result in a significant release of radioactive material to the environment or an increase in radiation levels. These severe accident conditions are associated with beyond-design-basis events, combinations of events for which the facility was not specifically designed. While these events could have consequences larger than those associated with design-basis events, their frequency is expected to be much lower than the design-basis event frequency. Natural phenomena (e.g., earthquake) and fire accidents creating a direct path for releases to the environment represent the situation with the most consequences to the public. Some types of accidents, such as procedure violations, spills of small quantities of material containing radioactive particles, and most other types of common human error occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material or radiation to result in significant release to the environment, although the impact to operational personnel may be as significant as that resulting from beyond-design-basis events. The airborne particles from a process-related accident would normally pass through at least one bank and possibly two to four banks of high-efficiency particulate air filters before entering the environment. Spent nuclear fuel handling operations are performed inside such confinement barriers as hot cells or canyon walls. The hot cells are equipped with significant safety features, such as an inert gas atmosphere, pressure control, and heat detection. These features are credited when their operability is not compromised by the sequence of events associated with the accident progression.

While severe accidents (also referred to as beyond-design-basis events) are expected to have the most significant impact, that is, the highest consequences, on the population, these accidents may not have as significant a risk impact on all receptors as higher-frequency, lower-consequence accidents. For this reason, higher-frequency accident scenarios were included in the accident analysis. Three categories of accidents were identified, and at least one accident scenario for each category was selected for analysis. The three categories consist of abnormal events (defined as events with a frequency of greater than 0.001 per year), design-basis events (frequencies between 1×10^{-3} and 1×10^{-6} per year), and beyond-design-basis events (frequencies less than 1×10^{-6}, but limited to those greater than 1×10^{-7} per year).

Based on review of the existing facility analyses and on guidance provided by the U.S. Department of Energy (DOE) in Section 6.9 of *Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements* (DOE 1993a), the following types of accidents were selected for each processing technology:

- Explosions
- Nuclear criticality
- Fire
- Earthquake
- Aircraft crash
- Spills/drops

Finally, no specific analyses of the results of terrorist or sabotage acts were considered. This is because the existing security measures in effect at the management sites would essentially preclude any sabotage or terrorist activity. In addition, any acts of terrorism would be expected to result in consequences that would be bounded by the results of the accident scenarios selected for detailed evaluation.

F.2.2.1 Accident Scenario Descriptions and Source Terms

This section describes the accident scenarios and corresponding source terms developed for ANL-W and SRS. The spectrum of accidents described below was used to determine the incremental consequences (public and worker doses) and risks associated with the treatment of sodium-bonded spent nuclear fuel at each site. These accident scenarios are consistent with those evaluated in either the facility safety analysis report, facility/site environmental reports, or various related DOE safety documents. Secondary accidents were considered when identified in the safety documents. The selected documents were identified and referenced in each of the accident scenarios described. When information was required to further clarify the accident condition, update some of the parameters, or facilitate the evaluation process, additional assumptions were made. Sometimes it was necessary to use different assumptions than those used in the referenced report; these are identified also. For example, under the proposed action of this EIS, the material at risk during an earthquake can be different than the materials considered in the facility safety analysis report. This change in assumption is necessary because the evaluations in this EIS focus only on the risk resulting from the implementation of alternatives (an incremental risk) and, therefore, address only the risk associated with the treatment of the sodium-bonded spent nuclear fuel. Cumulative risks can be determined by adding the incremental risks to the existing risks.

F.2.2.1.1 Source Terms

The source term (or building source term) is the amount of respirable radioactive material that is released to the air, in terms of curies or grams, assuming the occurrence of a postulated accident. The airborne source term is typically estimated by the following five-component linear equation:

$$\text{Source term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

where:

MAR	=	Material at Risk (grams or curies)
DR	=	Damage Ratio
ARF	=	Airborne Release Fraction (or Airborne Release Rate for continuous release)
RF	=	Respirable Fraction
LPF	=	Leak Path Factor

- *Material at Risk*—The material at risk is the amount of radionuclides (in curies of activity or grams for each radionuclide) available for release when acted upon by a given physical stress (i.e., an accident). The material at risk is specific to a given process in the facility of interest. It is not necessarily the total quantity of material present, but is that amount of material in the scenario of interest postulated to be available for release.

- *Damage Ratio*—This is the fraction of material exposed to the effects of the energy, force, or stress generated by the postulated event. For the accident scenarios discussed in this EIS, the value of the damage ratio varies from 0.0001 to 1.

- *Airborne Release Fraction*—This is the fraction of material that becomes airborne due to the accident. In this analysis, airborne release fraction values from the DOE Handbook on airborne release fractions are used (DOE 1994b).

- *Respirable Fraction*—This is the fraction of the material with a 10-micrometer (micron) or less aerodynamic-equivalent diameter particle size that could be retained in the respiratory system following inhalation. The respirable fraction values also are taken from the DOE Handbook on airborne release fractions (DOE 1994b).

- *Leak Path Factor*—The leak path factor accounts for the action of removal mechanisms (e.g., containment systems, filtration, deposition) to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces in the facility or the environment. A leak path factor of 1 (i.e., no reduction) is assigned in accident scenarios involving a major failure of confinement barriers.

F.2.2.1.2 Accident Scenario Descriptions and Source Terms at ANL-W

Accident Scenario Descriptions for Electrometallurgical Treatment Processing—The electrometallurgical treatment process would occur at the Fuel Conditioning Facility and the Hot Fuel Examination Facility at the ANL-W site. This process is detailed in Appendix C. The accident scenarios, identified in **Table F–2** and defined in the following paragraphs, are applicable to the electrometallurgical treatment process as proposed at ANL-W. This section also provides information addressing the material at risk and the various release fractions used to determine the source term for each accident selected for analysis.

Table F–2 Selected Accident Scenarios for Electrometallurgical Treatment Processing at ANL-W

Scenario	*Frequency (per year)*
Process-related spills/drops a. Salt powder spill b. Cask drop c. Salt transfer drop	 0.01 0.01 1×10^{-7}
Transuranic waste fire	0.001
Explosion	Not applicable
Design-basis earthquake	0.0002 [a] / 0.008 [b]
Aircraft crash	6×10^{-7} to 1×10^{-8}
Nuclear criticality	Less than 10^{-7}
Beyond-design-basis earthquake	0.00001

[a] At the Fuel Conditioning Facility.
[b] At the Hot Fuel Examination Facility.

Each accident scenario description sets the condition of the accident and provides a summary of material involved. As stated earlier, some of these accident scenarios are generic, but their applications are consistent with those evaluated in various ANL-W environmental and safety analyses. These scenarios include process-specific as well as generic and process-independent accidents. Tables F–3 through F–8 provide summaries of the accidents analyzed, the material at risk, and the release factors based on the fuel type expected to produce the most significant consequences, typically either Experimental Breeder Reactor-II (EBR-II) blanket or driver spent nuclear fuel, for each postulated accident.

- *Operational accident causing a salt powder spill in the Hot Fuel Examination Facility Main Cell*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into a final ceramic waste form. It is brought into the Hot Fuel Examination Facility in solid form and ground. The grinder is located in the Main Cell on a raised floor. In this accident scenario, it was assumed that during a transfer operation, the contents of a ground salt container are spilled into the pit beneath the floor. A portion of the salt powder becomes airborne and is carried through the ventilation system to the high-efficiency particulate air filters and released through the building stack. The release was assumed to occur over a one-hour period. The frequency of this accident was set at 0.01 per year, based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b).

  To estimate the fission product inventory in the electrorefiner salt, the option of not blending fuel types during electrorefining was used. The salt was assumed to come from the treatment of 5.56 metric tons of heavy metal of EBR-II blanket spent nuclear fuel elements (Goff et al. 1999b) or 1.1 metric tons of heavy metal of EBR-II driver spent nuclear fuel elements (Goff et al. 1999a), the point at which bulk replacement of salt in the electrorefiner is required either when the sodium limit is reached or when the treatment of each fuel type is completed. For the fuel types selected to represent the driver and blanket spent nuclear fuel, the fission product inventory in the salt would be conservative. Based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b), the material at risk was assumed to be 100 kilograms of ground salt containing the radionuclide concentrations as shown in **Table F–3**. Radionuclide distributions were developed for both EBR-II driver and blanket spent nuclear fuel. The radionuclide distributions for driver and blanket spent nuclear fuel are based on an average plutonium concentration in electrorefiner salt of 1.76 and 7.98 percent by weight, respectively (Goff et al. 1999a and 1999b). Portions of the spilled salt would become airborne. The maximum measured value for the 3-meter (10-foot) free-fall of dry cohesionless particles, with a mass median diameter of 1 to 2 microns, results in an airborne release fraction of 0.002 and a respirable fraction of 0.3 (DOE 1994b). The median particle size of the salt after grinding is approximately 200 microns, with only about 1 percent less than 20 microns in diameter (ANL 1999). The analysis, therefore, conservatively assumed that about 1 percent of the ground salt would have characteristics capable of resulting in the airborne release and respirable fractions identified above, resulting in a damage ratio of 0.01. The ventilation system and high-efficiency particulate air filters were assumed to function normally. The ventilation system consists of a two-stage high-efficiency particulate air filtration system were equivalent, with a first-stage high-efficiency particulate air filter efficiency of 99.9 percent and a second stage efficiency of 99 percent. Therefore, the leak path factor through the high-efficiency particulate air filters is 0.00001.

Table F–3 Material at Risk and Release Fraction Values for a Salt Powder Spill Accident at ANL-W

Material at Risk [a]			*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver* [b] *(curies)*					*Blanket*	*Driver*
Sr-90	580	35,000	0.01	0.002	0.3	0.00001	3.48×10^{-8}	2.10×10^{-6}
Y-90	580	35,000	0.01	0.002	0.3	0.00001	3.48×10^{-8}	2.10×10^{-6}
I-129	0.00104	0.0131	0.01	0.002	0.3	0.00001	6.24×10^{-14}	7.86×10^{-13}
Cs-134	9.63	313	0.01	0.002	0.3	0.00001	5.78×10^{-10}	1.88×10^{-8}
Cs-137	1,240	39,200	0.01	0.002	0.3	0.00001	7.44×10^{-8}	2.35×10^{-6}
Ba-137M	1,180	37,100	0.01	0.002	0.3	0.00001	7.08×10^{-8}	2.23×10^{-6}
Ce-144	45.1	526	0.01	0.002	0.3	0.00001	2.71×10^{-9}	3.16×10^{-8}
Pr-144	45.1	526	0.01	0.002	0.3	0.00001	2.71×10^{-9}	3.16×10^{-8}
Pm-147	292	14,700	0.01	0.002	0.3	0.00001	1.75×10^{-8}	8.82×10^{-7}
Sm-151	71.9	948	0.01	0.002	0.3	0.00001	4.31×10^{-9}	5.69×10^{-8}
Eu-154	5.28	101	0.01	0.002	0.3	0.00001	3.17×10^{-10}	6.06×10^{-9}
Eu-155	34.6	677	0.01	0.002	0.3	0.00001	2.08×10^{-9}	4.06×10^{-8}
Th-228	0.000111	0.0091	0.01	0.002	0.3	0.00001	6.66×10^{-15}	5.48×10^{-13}
Np-237	0.00602	0.0513	0.01	0.002	0.3	0.00001	3.61×10^{-13}	3.08×10^{-12}
Pu-238	6.44	66.8	0.01	0.002	0.3	0.00001	3.86×10^{-10}	4.01×10^{-9}
Pu-239	517	108	0.01	0.002	0.3	0.00001	3.10×10^{-8}	6.48×10^{-9}
Pu-240	35.5	3.67	0.01	0.002	0.3	0.00001	2.13×10^{-9}	2.20×10^{-10}
Pu-241	144	8.93	0.01	0.002	0.3	0.00001	8.64×10^{-9}	5.36×10^{-10}
Am-241	11.7	0.0694	0.01	0.002	0.3	0.00001	7.02×10^{-10}	4.16×10^{-12}
Am-242M	0.121	0.0000588	0.01	0.002	0.3	0.00001	7.26×10^{-12}	3.53×10^{-15}

[a] Radionuclide inventory from Appendix D.

[b] Use of data contained in the draft report (Goff et al. 1999a) for the driver spent nuclear fuel results in higher material-at-risk values for most isotopes presented in Table F–3 compared to data in the final report (Goff et al. 1999b). Therefore, these material-at- risk estimates were not revised to reflect data in the final report.

- *Cask drop and gaseous fission product release*—Spent nuclear fuel casks would be handled frequently when the sodium-bonded fuel is processed. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) Spent nuclear fuel stored in the Radioactive Scrap and Waste Facility would be transferred to the Fuel Conditioning Facility for processing. Spent nuclear fuel would be received from off site at the Hot Fuel Examination Facility and would be transferred to the Fuel Conditioning Facility for processing. The HFEF-5 cask would be used to move EBR-II driver and blanket spent nuclear fuel from the Radioactive Scrap and Waste Facility to the Fuel Conditioning Facility. The postulated accident is described in the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b). The accident involves a cask dropped during unloading, resulting in seal and fuel cladding failure sufficient to release gaseous and volatile fission products to the atmosphere. The drop could be initiated by failure of lifting equipment, slings, hooks, cables, or human error by the lifting equipment operator. The cask drop was assumed conservatively to result in an unfiltered release of gaseous and volatile fission products. The release was assumed to be a puff release at ground level. Dropping of casks, while rare, is nevertheless categorized as anticipated, since such events have happened in the past and may be expected to occur over the lifetime of the facility. The frequency of cask dropping was assumed to be 0.01 per year, consistent with that used in the Safety Analysis Report for the Hot Fuel Examination Facility.

The HFEF-5 cask can contain two EBR-II driver spent nuclear fuel assemblies. It was assumed conservatively that the equivalent of one assembly (61 elements) fails in the accident. The material at risk, as shown in **Table F–4**, would be the equivalent of one EBR-II driver or blanket spent nuclear fuel assembly. The damage ratio for the failed elements was assumed to be 1, since all gaseous and volatile fission products conservatively could be released to the cask following cladding failure. The airborne release and respirable fractions for gases were each assumed to be 1, and 1×10^{-7} for cesium from the dislodgement of surface contamination (DOE 1995a). The accident was assumed to occur outdoors, with a leak path factor of 1.

Table F–4 Material at Risk and Release Fraction Values for a Cask Drop Accident at ANL-W

Material at Risk[a]			*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
H-3 [b]	0.335	5.17	1	1	1	1	0.335	5.17
Kr-85	2.44	79.4	1	1	1	1	2.44	79.4
Cs-134	0.63	7.39	1	1.0×10^{-7}	1	1	6.30×10^{-8}	7.39×10^{-7}
Cs-137	81.3	928	1	1.0×10^{-7}	1	1	8.13×10^{-6}	0.0000928

[a] Data for one assembly based on Appendix D curie content data.
[b] It was assumed that 1 percent of this release becomes oxidized.

- *Salt transfer drop during movement from the Fuel Conditioning Facility to the Hot Fuel Examination Facility*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into a final ceramic waste form. It is transferred in large chunks within the HFEF-5 cask. Transfer is via forklift or truck. In this scenario, a severe vehicle accident occurs, resulting in a breach of the inner and outer salt container. The accident could be caused by operator error or equipment failure. The accident is considered beyond-design-basis because of the durability of the shielded HFEF-5 canister. There would be over 200 transfers of salt from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. A probability of 1×10^{-7} was assumed. The release occurs at ground level with a duration of one hour.

  Table F–5 provides the isotopic material at risk for a total material at risk of 20 kilograms of salt. The salt is in chunks (i.e., ice cube-size) and is not combustible. No significant release was assumed from the large pieces. Some of the salt pieces would experience brittle fracture and release particulates. A brittle fracture particulate fraction for solidified salt would be 0.0001 for particles less than 10 microns in diameter (ANL 1998b); therefore, a damage ratio of 0.0001 was assumed. Conservatively, the same airborne release fraction and respirable fraction values were used for this scenario as for the salt powder spill in the Hot Fuel Examination Facility Main Cell; that is, the airborne release fraction for powder is 0.002 and the respirable fraction is 0.3 (DOE 1994b). The accident occurs outdoors; therefore, the leak path factor is 1.

Table F–5 Material at Risk and Release Fraction Values for a Salt Transfer Drop Accident at ANL-W

Material at Risk [a]							*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Blanket*	*Driver*
Sr-90	116	7,000	0.0001	0.002	0.3	1	6.96×10^{-6}	0.000420
Y-90	116	7,000	0.0001	0.002	0.3	1	6.96×10^{-6}	0.000420
I-129	0.000207	0.00261	0.0001	0.002	0.3	1	1.24×10^{-11}	1.57×10^{-10}
Cs-134	1.92	62.5	0.0001	0.002	0.3	1	1.15×10^{-7}	3.75×10^{-6}
Cs-137	249	7,850	0.0001	0.002	0.3	1	0.0000149	0.000471
Ba-137M	236	7,420	0.0001	0.002	0.3	1	0.0000142	0.000445
Ce-144	9.02	105	0.0001	0.002	0.3	1	5.41×10^{-7}	6.30×10^{-6}
Pr-144	9.02	105	0.0001	0.002	0.3	1	5.41×10^{-7}	6.30×10^{-6}
Pm-147	58.5	2,930	0.0001	0.002	0.3	1	3.51×10^{-6}	0.000176
Sm-151	14.4	190	0.0001	0.002	0.3	1	8.64×10^{-7}	0.0000114
Eu-154	1.06	20.1	0.0001	0.002	0.3	1	6.36×10^{-8}	1.21×10^{-6}
Eu-155	6.91	135	0.0001	0.002	0.3	1	4.15×10^{-7}	8.10×10^{-6}
Th-228	0.0000223	0.00183	0.0001	0.002	0.3	1	1.34×10^{-8}	1.10×10^{-10}
Np-237	0.00120	0.0103	0.0001	0.002	0.3	1	7.20×10^{-11}	6.18×10^{-10}
Pu-238	1.29	13.4	0.0001	0.002	0.3	1	7.74×10^{-8}	8.04×10^{-7}
Pu-239	103	21.6	0.0001	0.002	0.3	1	6.18×10^{-6}	1.30×10^{-6}
Pu-240	7.11	0.733	0.0001	0.002	0.3	1	4.27×10^{-7}	4.40×10^{-8}
Pu-241	28.8	1.79	0.0001	0.002	0.3	1	1.73×10^{-6}	1.07×10^{-7}
Am-241	2.34	0.0139	0.0001	0.002	0.3	1	1.40×10^{-7}	8.34×10^{-10}
Am-242M	0.0243	0.0000118	0.0001	0.002	0.3	1	1.46×10^{-9}	7.08×10^{-13}

[a] The material at risk is the isotope in 20 kilograms of salt, which is 20 percent of the values given in Table F-3.

- *Transuranic waste fire*—Transuranic waste is generated as a result of treatment operations, as well as other operations, at ANL-W. This waste is placed in containers and temporarily stored (staged) at ANL-W pending shipment to the Radioactive Waste Management Complex. A fire was postulated to occur in a 1.2 × 1.2 × 2.4-meter (4 × 4 × 8-foot) solid transuranic waste box because of spontaneous combustion, pyrophoric material, vehicle accident, electrical failure, or poor housekeeping. The fire consumes the contents of one box of transuranic waste. The accident was assumed to occur outdoors during handling. The release occurs at ground level over one hour. The Final Safety Analysis Report for the Fuel Conditioning Facility assigned an accident frequency in the range of 0.0001 to 0.01 (ANL 1998a). Here, the accident was assumed to have a frequency of 0.001 per year.

  The material at risk, as shown in **Table F–6**, was assumed to be one box of transuranic waste. The waste boxes are loaded with 1/20th of 0.34 curies of alpha activity, as described in the Fuel Conditioning Facility Final Safety Analysis Report (ANL 1998a). The material at risk is 0.017 curies of transuranic nuclides, with the nuclide distribution associated with the generic contents of a transuranic waste container. The damage ratio was assumed to be 1, since all waste in the container was assumed to be involved in the fire. An airborne release fraction of 0.0005 and a respirable fraction of 1 for burning of surface contaminated waste was used (DOE 1994b). The leak path factor was assumed to be 1. No credit was taken for building confinement.

Table F–6 Material at Risk and Release Fraction Values for a Transuranic Waste Fire Accident at ANL-W

Material at Risk [a]		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Isotope	*Curies*					
Pu-238	0.000153	1	0.0005	1	1	7.67×10^{-8}
Pu-239	0.0123	1	0.0005	1	1	6.15×10^{-6}
Pu-240	0.000846	1	0.0005	1	1	4.23×10^{-7}
Pu-241	0.00343	1	0.0005	1	1	1.72×10^{-6}
Am-241	0.000266	1	0.0005	1	1	1.33×10^{-7}

[a] The material at risk is for a generic waste package, not for any specific spent nuclear fuel.

- *Design-basis earthquake - multifacility effects*—In the Fuel Conditioning Facility, the argon cell contains the equipment for processing sodium-bonded spent nuclear fuel into salt and metallic waste forms and a uranium metal product. All operations involving bare fuel are conducted in the argon cell because the inert atmosphere precludes pyrophoric metal fire. Fire cannot occur unless sufficient oxygen enters the cell through a cell breach. The walls, ceiling, and floor of the argon cell are constructed from reinforced concrete with thicknesses ranging from 1.2 to 1.5 meters (4 to 5 feet). It also has a gas-tight steel lining. It was assumed that the accident occurs during electrometallurgical treatment operations. Chopped fuel, electrorefiner salts, cathodes, and anodes are all present in the argon cell. Consistent with the assumption given in the Fuel Conditioning Facility Safety Analysis Report, a design-basis earthquake at this facility would result in a cell breach and in-leakage of air. The air in the cell would cause pyrophoric metals to ignite and burn. The Final Safety Analysis Report for the Fuel Conditioning Facility (ANL 1998a) identifies the seismic design goal for the facility to be the ability to withstand a 0.21 g design-basis earthquake. This event is identified as having a return frequency of 0.0002 per year. At this earthquake level, the electrorefiners are seismically qualified, and no spill of molten salt would occur. The safety exhaust system also would remain operational, although breaches could occur in the argon cell boundary after a design-basis earthquake. The safety exhaust building, which includes the high-efficiency particulate air filters, is designed to withstand an earthquake of 0.24 g, and was assumed to function as designed, filtering the cell atmosphere prior to release through the Fuel Conditioning Facility stack.

  In the Hot Fuel Examination Facility, grinding of salt into powder was assumed to be occurring in the Main Cell. The grinder is located in the Hot Fuel Examination Facility Main Cell on a raised floor consisting of steel plates resting on supports. Underneath the floor is a 2.4-meter-deep (8-foot-deep) pit that houses the ventilation ductwork and high-efficiency particulate air filters. At the Hot Fuel Examination Facility, a design-basis earthquake was assumed to cause the vessel containing ground salt to topple and the powder to spill out. Since the ventilation system was not seismically qualified, it was assumed to fail and result in an unfiltered release. It also was assumed that the design-basis earthquake would cause a loss of electrical power, which would result in failure of the ventilation system. The Main Cell breaches at piping or ventilation penetrations, providing a release path for the suspended powder. The releases occur over a one-hour period, and were modeled as a ground-level release.

  The Hot Fuel Examination Facility has been analyzed for a 0.14 g design-basis earthquake, an event with a return frequency of 0.001 per year and a performance goal of 0.0001 per year. The functionality of equipment after a 0.14 g earthquake has not been determined as yet. However, all major systems remained functional during the 0.03 g Borah earthquake in 1983, an event with a return frequency of 0.008 per year. While it is expected that the equipment would survive a 0.14 g earthquake with a frequency of 0.001 per year, the 0.008 per year earthquake frequency (ANL 1998b) was used conservatively to represent the upper bound of the design-basis earthquake, which would result in a salt powder spill and the failure of the ventilation system. This frequency is nearly two orders of magnitude higher than that corresponding to a 0.21 g earthquake that could impact both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Therefore, 0.008 per year was used for the design-basis earthquake accident frequency for the Hot Fuel Examination Facility.

In the Fuel Conditioning Facility, the material at risk is chopped spent nuclear fuel and uranium in two electrorefiner cathodes in the argon cell at the time of the accident. **Table F–7** provides material-at-risk values for the isotopes of concern. The bounding inventory is 20 kilograms (44 pounds) of chopped fuel and the uranium in two solid electrorefiner cathodes (ANL 1998a). The solid cathodes contain 17 kilograms (37 pounds) of uranium. (Uranium is considered a toxic chemical in the chemical accident assessment, Section F.3.) The material at risk is, therefore, the 20 kilograms (44 pounds) of chopped spent nuclear fuel. For the metal fire in the argon cell, the damage ratio was assumed to be 1, since all materials in the material at risk are released to the cells in the accident. For the Fuel Conditioning Facility, the airborne release fraction values are 1 for krypton-85, 0.00025 for cesium, and 2.5×10^{-6} for strontium, uranium, and the transuranic waste nuclides; the respirable fractions are each 1 (DOE 1995a). For the Fuel Conditioning Facility, the safety exhaust system remains functional, and the release is filtered through high-efficiency particulate air filters. A leak path factor of 0.00001 was assumed for all particulates.

Table F–7 Material at Risk and Release Fraction Values for a Design-Basis Earthquake at ANL-W

Accident	*Material at Risk* [a]			*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
	Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Design-basis earthquake and salt powder spill at the Hot Fuel Examination Facility	Sr-90	580	35,000	0.01	0.0020	0.30	0.125	0.000435	0.0263
	Y-90	580	35,000	0.01	0.0020	0.30	0.125	0.000435	0.0263
	I-129	0.00104	0.0131	0.01	0.0020	0.30	0.125	7.80×10^{-10}	9.83×10^{-9}
	Cs-134	9.63	313	0.01	0.0020	0.30	0.125	7.22×10^{-6}	0.000235
	Cs-137	1,240	39,200	0.01	0.0020	0.30	0.125	0.000930	0.0294
	Ba-137M	1,180	37,100	0.01	0.0020	0.30	0.125	0.000885	0.0278
	Ce-144	45.1	526	0.01	0.0020	0.30	0.125	0.0000338	0.000395
	Pr-144	45.1	526	0.01	0.0020	0.30	0.125	0.0000338	0.000395
	Pm-147	292	14,700	0.01	0.0020	0.30	0.125	0.000219	0.0110
	Sm-151	71.9	948	0.01	0.0020	0.30	0.125	0.0000539	0.000711
	Eu-154	5.28	101	0.01	0.0020	0.30	0.125	3.96×10^{-6}	0.0000758
	Eu-155	34.6	677	0.01	0.0020	0.30	0.125	0.0000260	0.000508
	Th-228	0.000111	0.00913	0.01	0.0020	0.30	0.125	8.33×10^{-11}	6.85×10^{-9}
	Np-237	0.00602	0.0513	0.01	0.0020	0.30	0.125	4.51×10^{-9}	3.85×10^{-8}
	Pu-238	6.44	66.8	0.01	0.0020	0.30	0.125	4.83×10^{-6}	0.0000501
	Pu-239	517	108	0.01	0.0020	0.30	0.125	0.000388	0.0000810
	Pu-240	35.5	3.67	0.01	0.0020	0.30	0.125	0.0000266	2.75×10^{-6}
	Pu-241	144	8.93	0.01	0.0020	0.30	0.125	0.000108	6.70×10^{-6}
	Am-241	11.7	0.0694	0.01	0.0020	0.30	0.125	8.78×10^{-6}	5.21×10^{-8}
	Am-242M	0.121	0.0000588	0.01	0.0020	0.30	0.125	9.08×10^{-8}	4.41×10^{-11}
Design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell	H-3	0.142	24.4	1	1	1	1	0.142	24.4
	C-14	0.00119	3,980	1	2.5×10^{-6}	1	0.00001	2.99×10^{-14}	9.95×10^{-8}
	Fe-55	1.80	97.4	1	2.5×10^{-6}	1	0.00001	4.51×10^{-11}	2.44×10^{-9}
	Co-60	0.318	9.62	1	2.5×10^{-6}	1	0.00001	7.95×10^{-12}	2.41×10^{-10}
	Ni-63	0.0612	4.58	1	2.5×10^{-6}	1	0.00001	1.53×10^{-12}	1.15×10^{-10}
	Kr-85	1.04	378	1	1	1	1	1.04	378
	Sr-90	16.1	3,940	1	2.5×10^{-6}	1	0.00001	4.04×10^{-10}	9.85×10^{-8}
	Y-90	16.1	3,940	1	2.5×10^{-6}	1	0.00001	4.04×10^{-10}	9.85×10^{-8}
	Ru-106	2.70	30.2	1	0.00025	1	0.00001	6.75×10^{-9}	7.55×10^{-8}
	Rh-106	2.70	30.2	1	2.5×10^{-6}	1	0.00001	6.75×10^{-11}	7.55×10^{-10}

Accident	Isotope	Material at Risk [a] Blanket (curies)	Material at Risk [a] Driver (curies)	Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies) Blanket	Source Term (curies) Driver
Design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell (cont'd)	Cd-113M	0.0142	0.928	1	2.5×10^{-6}	1	0.00001	3.56×10^{-13}	2.32×10^{-11}
	Sb-125	0.462	59.2	1	2.5×10^{-6}	1	0.00001	1.16×10^{-11}	1.48×10^{-9}
	Te-125M	0.190	24.6	1	2.5×10^{-6}	1	0.00001	4.76×10^{-12}	6.15×10^{-10}
	I-129	0.0000288	0.00147	1	1	1	1	0.0000288	0.00147
	Cs-134	0.268	35.2	1	0.00025	1	0.00001	6.70×10^{-10}	8.80×10^{-8}
	Cs-137	34.6	4,420	1	0.00025	1	0.00001	8.65×10^{-8}	0.0000111
	Ba-137M	32.8	4,180	1	2.5×10^{-6}	1	0.00001	8.20×10^{-10}	1.05×10^{-7}
	Ce-144	1.25	59.2	1	2.5×10^{-6}	1	0.00001	3.14×10^{-11}	1.48×10^{-9}
	Pr-144	1.25	59.2	1	2.5×10^{-6}	1	0.00001	3.14×10^{-11}	1.48×10^{-9}
	Pm-147	8.14	1,650	1	2.5×10^{-6}	1	0.00001	2.04×10^{-10}	4.13×10^{-8}
	Sm-151	2.00	107	1	2.5×10^{-6}	1	0.00001	5.00×10^{-11}	2.67×10^{-9}
	Eu-154	0.147	11.3	1	2.5×10^{-6}	1	0.00001	3.67×10^{-12}	2.84×10^{-10}
	Eu-155	0.962	76.2	1	2.5×10^{-6}	1	0.00001	2.41×10^{-11}	1.91×10^{-9}
	Th-228	3.10×10^{-6}	0.00103	1	2.5×10^{-6}	1	0.00001	7.75×10^{-17}	2.57×10^{-14}
	U-234	0.0000266	0.808	1	2.5×10^{-6}	1	0.00001	6.65×10^{-16}	2.02×10^{-11}
	U-235	0.0000754	0.0262	1	2.5×10^{-6}	1	0.00001	1.89×10^{-15}	6.55×10^{-13}
	U-236	0.0000848	0.0242	1	2.5×10^{-6}	1	0.00001	2.12×10^{-15}	6.05×10^{-13}
	U-238	0.00654	0.00222	1	2.5×10^{-6}	1	0.00001	1.64×10^{-13}	5.55×10^{-14}
	Np-237	2.60×10^{-6}	0.00578	1	2.5×10^{-6}	1	0.00001	6.50×10^{-17}	1.45×10^{-13}
	Pu-238	0.188	3.32	1	2.5×10^{-6}	1	0.00001	4.70×10^{-12}	8.30×10^{-11}
	Pu-239	15.1	5.38	1	2.5×10^{-6}	1	0.00001	3.77×10^{-10}	1.35×10^{-10}
	Pu-240	1.04	0.182	1	2.5×10^{-6}	1	0.00001	2.59×10^{-11}	4.56×10^{-12}
	Pu-241	4.20	0.444	1	2.5×10^{-6}	1	0.00001	1.05×10^{-10}	1.11×10^{-11}
	Am-241	0.326	0.00782	1	2.5×10^{-6}	1	0.00001	8.15×10^{-12}	1.96×10^{-13}
	Am-242M	0.00338	6.62×10^{-6}	1	2.5×10^{-6}	1	0.00001	8.45×10^{-14}	1.66×10^{-16}

[a] Radionuclide inventory from Appendix D.

During the postulated event, 100 kilograms (220 pounds) of solidified salt powder with the same concentration of radionuclides as described previously for the salt powder spill accident are assumed to be spilled in the Hot Fuel Examination Facility Main Cell. As previously discussed, in a salt powder spill, less than 1 percent of the salt would have the characteristics capable of resulting in an airborne release, i.e., a damage ratio of 0.01 was used. For the powder spill within the cell, an airborne release fraction of 0.002 and a respirable fraction of 0.3 were assumed (DOE 1994b). These are the same values as those used for the salt powder spill accident described previously. The Hot Fuel Examination Facility leak path for the release is through three enclosures before reaching the outside: the Main Cell, ducts and pipes, and the building. Consistent with the facility safety analysis report assumption, a leak path factor of 0.5 was assigned to each enclosure for plate-out and settling of the airborne powder. Therefore, the total leak path factor is $0.5 \times 0.5 \times 0.5 = 0.125$.

- *Aircraft crash*—The potential for an aircraft crash was evaluated. The methodology for evaluating the likelihood of an aircraft crash is documented in the *DOE Standard: Accident Analysis for Aircraft Crash into Hazardous Facilities* (DOE 1996c). At Idaho National Engineering and Environmental Laboratory (INEEL), the probabilities of a small and large aircraft crash are 2.3×10^{-4} and 1.0×10^{-6} crashes per square kilometer (9×10^{-5} and 4×10^{-7} crashes per square mile) per year, respectively. Using guidance in this DOE standard, the effective area of the Fuel Conditioning Facility was calculated accounting for aircraft wing span and potential skid distance. The effective area of the Fuel Conditioning Facility is about 0.078 square kilometers (0.03 square miles) for a large aircraft, and 0.018 square kilometers (0.007 square miles) for a small aircraft. The effective area of the Fuel Conditioning Facility is conservative because the combined area of the air and argon cells, where the hazardous materials are contained, is smaller than the total area of the building. Multiplying the effective area by INEEL-specific crash rates gives an estimated probability of a crash into the Fuel Conditioning Facility of 1×10^{-8} for large aircraft and 6×10^{-7} for small aircraft. Comparable probabilities are applicable to the Hot Fuel Examination Facility. A large aircraft crash is not reasonably foreseeable, and given the 1.2 to 1.5-meter-thick (4 to 5-foot-thick) walls of the cells and the "buffer" provided by the building exterior walls, the crash of a small aircraft is unlikely to result in any damage to the cells. Damage from the more probable seismic events analyzed is considered to bound the damage that could result from a small aircraft crash. Also, seismic events affect more than one facility, while an aircraft crash could affect only one facility. Therefore, an aircraft crash was not analyzed separately.

- *Nuclear criticality*—The potential for a nuclear criticality was considered in the accident analysis. Nuclear criticality has been evaluated in the safety analyses documented for the ANL-W facilities, as required by DOE. The existing safety analyses conclude that nuclear criticality is beyond the design-basis of the facilities proposed for the electrometallurgical treatment alternative and, therefore, has a probability of less than 1×10^{-6} per year. This conclusion is based on a lack of nuclear moderator materials, equipment design, and inventory controls, as well as numerous other administrative controls and operating procedures. The intent of the process is to dilute, rather than concentrate, fissile materials. Fuel storage racks and processing equipment are designed to maintain their safety function during the design-basis earthquake. Even in a beyond-design-basis earthquake (maximum frequency of 0.00001 per year), nuclear materials would have to come together in an ideal critical array for criticality to be possible. For example, it would require more than the equivalent of 10 EBR-II driver spent nuclear fuel assemblies (610 individual elements) in an ideal geometric configuration to create a potential criticality hazard. During processing, some fuel would be stored in the hot cells. This fuel is stored in the storage cans within the floor storage pits. The floor storage pits are evenly spaced 61 centimeters (2 feet) from the center, located almost entirely on a 3-meter-thick (10-foot-thick) hot cell concrete floor. These pits are designed to maintain criticality-safe configurations under all normal and design-basis abnormal conditions, including a design-basis earthquake (ANL 1998a). An evaluation of earthquake loading has concluded that no uplifting of the hot cell floor would occur in a beyond-design-basis earthquake of 0.3 g peak ground acceleration (corresponding to an earthquake frequency of 0.00001 per year) (ANL 1999). Therefore, the conditional probability of creating a criticality hazard configuration, given a beyond-design-basis earthquake, was estimated to be no greater than 0.01. Therefore, criticality is not considered to be reasonably foreseeable, and was not analyzed quantitatively.

- *Beyond-design-basis earthquake*—This scenario is similar to the design-basis earthquake except that the safety exhaust system was not assumed to function at the Fuel Conditioning Facility, and an electrorefiner was assumed to spill its molten salt. Also, since spent nuclear fuel elements are stored in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, a fraction of stored fuel elements were assumed to experience cladding failure and release of gaseous and volatile fission products. All releases were modeled as ground-level releases. The Fuel Conditioning Facility horizontal acceleration design-basis is 0.21 g, and the newer safety equipment building is designed for a 0.24 g horizontal acceleration. A 0.24 g peak acceleration corresponds to an earthquake frequency at ANL-W of approximately 0.0001 per year (WCFS 1998). The Fuel Conditioning Facility natural phenomena hazard performance goal is a frequency of 0.00001 (DOE 1994a). (The Hot Fuel Examination Facility performance goal is 0.0001.) The performance goal can be interpreted as the frequency level at which facility damage will initiate. The Fuel Conditioning Facility and safety exhaust system are not expected to suffer damage from earthquakes with frequencies higher than this. Therefore, the upper bound for the beyond-design-basis earthquake frequency was assumed to correspond to the frequency of the performance goal, 0.00001per year.

  The material at risk, provided in **Table F–8**, would be the same as for the design-basis earthquake, with the addition of the salt in the electrorefiners and the fuel elements and subassemblies in storage. Although the electrorefiners are seismically qualified, one of the two electrorefiners in the Fuel Conditioning Facility argon cell was assumed conservatively to spill its molten salt. It was assumed that approximately 700 kilograms (1,540 pounds) of salt are fully loaded with radionuclides from the processing of 5.56 metric tons of heavy metal of blanket spent nuclear fuel elements or 1.1 metric tons of driver spent nuclear fuel elements and are about to be replaced at the time of the accident. The damage ratio for all but the fuel assemblies in storage was assumed to be 1, as in the design-basis earthquake. Both the blanket and driver spent nuclear fuel elements are stored in racks with the cladding intact. In the earthquake, some could be expected to fall out of the racks or be hit by falling debris, but it is not reasonable to assume all assemblies would be damaged. It was assumed that 10 percent of the elements stored in the cells at the time of the earthquake experience cladding failure and release gaseous and volatile fission products. For the driver spent nuclear fuel elements, this is the equivalent of 12 driver assemblies (or 50 kilograms [110 pounds] of heavy metal). Ten percent of the stored blanket elements is the equivalent of 370 kilograms (771 pounds) of heavy metal. The airborne release fraction and respirable fraction values are the same as for the design-basis earthquake, with the addition of krypton and cesium from the failed EBR-II driver and blanket fuel. The airborne release fraction and respirable fraction values for krypton and tritium (H-3), both elements in the gaseous state, are each 1. For the molten salt spill, the airborne release fraction and respirable fraction values for viscous solutions (DOE 1994b) were used: 4×10^{-6} (0.0004 for iodine and cesium) for the airborne release fraction and 0.8 for the respirable fraction. The forces associated with the beyond-design-basis earthquake were assumed to result in the failure of confinement integrity. The cells were assumed to experience major failure, and the release would be directly to the atmosphere. The leak path factor is 1.

Table F–8 Material at Risk and Release Fraction Values for a Beyond-Design-Basis Earthquake at ANL-W

Accident	*Material at Risk*[a]			*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
	Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Beyond-design-basis earthquake and salt powder spill in the Hot Fuel Examination Facility	Sr-90	580	35,000	0.01	0.002	0.3	1	0.00348	0.21
	Y-90	580	35,000	0.01	0.002	0.3	1	0.00348	0.21
	I-129	0.00104	0.0131	0.01	0.002	0.3	1	6.24×10^{-9}	7.86×10^{-8}
	Cs-134	9.63	313	0.01	0.002	0.3	1	0.0000578	0.00188
	Cs-137	1,240	39,200	0.01	0.002	0.3	1	0.00744	0.235
	Ba-137M	1,180	37,100	0.01	0.002	0.3	1	0.00708	0.223
	Ce-144	45.1	526	0.01	0.002	0.3	1	0.000271	0.00316
	Pr-144	45.1	526	0.01	0.002	0.3	1	0.000271	0.00316
	Pm-147	292	14,700	0.01	0.002	0.3	1	0.00175	0.0882
	Sm-151	71.9	948	0.01	0.002	0.3	1	0.000431	0.00569
	Eu-154	5.28	101	0.01	0.002	0.3	1	0.0000317	0.000606
	Eu-155	34.6	677	0.01	0.002	0.3	1	0.000148	0.00406
	Th-228	0.000111	0.00913	0.01	0.002	0.3	1	6.66×10^{-10}	5.48×10^{-8}
	Np-237	0.00602	0.0513	0.01	0.002	0.3	1	3.61×10^{-8}	3.08×10^{-7}
	Pu-238	6.44	66.8	0.01	0.002	0.3	1	0.0000386	0.000401
	Pu-239	517	108	0.01	0.002	0.3	1	0.00310	0.000648
	Pu-240	35.5	3.67	0.01	0.002	0.3	1	0.000213	0.000022
	Pu-241	144	8.93	0.01	0.002	0.3	1	0.000864	0.0000536
	Am-241	11.7	0.0694	0.01	0.002	0.3	1	0.0000702	4.16×10^{-7}
	Am-242M	0.121	0.0000588	0.01	0.002	0.3	1	7.26×10^{-7}	3.53×10^{-10}
Beyond-design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell	H-3	0.142	24.4	1	1	1	1	0.142	24.4
	C-14	0.00119	3,980	1	2.5×10^{-6}	1	1	2.99×10^{-9}	0.00995
	Fe-55	1.80	97.4	1	2.5×10^{-6}	1	1	4.51×10^{-6}	0.000244
	Co-60	0.318	9.62	1	2.5×10^{-6}	1	1	7.95×10^{-7}	0.0000241
	Ni-63	0.0612	458	1	2.5×10^{-6}	1	1	1.53×10^{-7}	0.00115
	Kr-85	1.04	378	1	1	1	1	1.04	378
	Sr-90	16.1	3,940	1	2.5×10^{-6}	1	1	0.0000404	0.00985
	Y-90	16.1	3,940	1	2.5×10^{-6}	1	1	0.0000404	0.00985
	Ru-106	2.70	30.2	1	0.00025	1	1	0.000675	0.00755

Accident	Material at Risk[a]			Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies)	
	Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
Beyond-design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell (cont'd)	Rh-106	2.70	30.2	1	2.5×10^{-6}	1	1	6.75×10^{-6}	0.0000755
	Cd-113M	0.0142	0.928	1	2.5×10^{-6}	1	1	3.56×10^{-8}	2.32×10^{-6}
	Sb-125	0.462	59.2	1	2.5×10^{-6}	1	1	1.15×10^{-6}	0.000148
	Te-125M	0.190	24.6	1	2.5×10^{-6}	1	1	4.76×10^{-7}	0.0000615
	I-129	0.0000288	0.00147	1	1	1	1	0.0000288	0.00147
	Cs-134	0.268	35.2	1	0.00025	1	1	0.000067	0.00880
	Cs-137	34.6	4,420	1	0.00025	1	1	0.00865	1.10
	Ba-137M	32.8	4,180	1	2.5×10^{-6}	1	1	0.000082	0.0105
	Ce-144	1.25	59.2	1	2.5×10^{-6}	1	1	3.14×10^{-6}	0.000148
	Pr-144	1.25	59.2	1	2.5×10^{-6}	1	1	3.14×10^{-6}	0.000148
	Pm-147	8.14	1,650	1	2.5×10^{-6}	1	1	0.0000204	0.00413
	Sm-151	2.00	107	1	2.5×10^{-6}	1	1	5.00×10^{-6}	0.000267
	Eu-154	0.147	11.3	1	2.5×10^{-6}	1	1	3.67×10^{-7}	0.0000284
	Eu-155	0.962	76.2	1	2.5×10^{-6}	1	1	2.41×10^{-6}	0.000191
	Th-228	3.10×10^{-6}	0.00103	1	2.5×10^{-6}	1	1	7.75×10^{-12}	2.57×10^{-9}
	Np-237	0.0000266	0.808	1	2.5×10^{-6}	1	1	6.65×10^{-11}	2.02×10^{-6}
	U-234	0.0000754	0.0262	1	2.5×10^{-6}	1	1	1.89×10^{-10}	6.55×10^{-8}
	U-235	0.0000848	0.0242	1	2.5×10^{-6}	1	1	2.12×10^{-10}	6.05×10^{-8}
	U-236	0.00654	0.00222	1	2.5×10^{-6}	1	1	1.64×10^{-8}	5.55×10^{-9}
	U-238	2.60×10^{-6}	0.00578	1	2.5×10^{-6}	1	1	6.50×10^{-12}	1.45×10^{-8}
	Pu-238	0.188	3.32	1	2.5×10^{-6}	1	1	4.70×10^{-7}	8.30×10^{-6}
	Pu-239	15.1	5.38	1	2.5×10^{-6}	1	1	0.0000377	0.0000135
	Pu-240	1.04	0.182	1	2.5×10^{-6}	1	1	2.59×10^{-6}	4.56×10^{-7}
	Pu-241	4.20	0.444	1	2.5×10^{-6}	1	1	0.0000105	1.11×10^{-6}
	Am-241	0.326	0.00782	1	2.5×10^{-6}	1	1	8.15×10^{-7}	1.96×10^{-8}
	Am-242M	0.00338	6.62×10^{-6}	1	2.5×10^{-6}	1	1	8.45×10^{-9}	1.66×10^{-11}
Beyond-design-basis earthquake and liquid salt spill at the Fuel Conditioning Facility	Sr-90	4,490	245,000	1	4.0×10^{-6}	0.8	1	0.0144	0.784
	Y-90	4,490	245,000	1	4.0×10^{-6}	0.8	1	0.0144	0.784
	I-129	0.00801	0.0917	1	0.0004	0.8	1	2.56×10^{-6}	0.0000293
	Cs-134	74.5	2,190	1	0.0004	0.8	1	0.0238	0.701
	Cs-137	9,620	274,000	1	0.0004	0.8	1	3.08	87.8
	Ba-137M	9,120	260,000	1	4.0×10^{-6}	0.8	1	0.0292	0.831
	Ce-144	349	3,680	1	4.0×10^{-6}	0.8	1	0.00112	0.0118
	Pr-144	349	3,680	1	4.0×10^{-6}	0.8	1	0.00112	0.0118
	Pm-147	2,260	103,000	1	4.0×10^{-6}	0.8	1	0.00723	0.329
	Sm-151	556	6,640	1	4.0×10^{-6}	0.8	1	0.00178	0.0212
	Eu-154	40.8	707	1	4.0×10^{-6}	0.8	1	0.000131	0.00226
	Eu-155	267	4,740	1	4.0×10^{-6}	0.8	1	0.000854	0.0152
	Th-228	0.000862	0.0639	1	4.0×10^{-6}	0.8	1	2.76×10^{-9}	2.05×10^{-7}
	Np-237	0.0465	0.359	1	4.0×10^{-6}	0.8	1	1.49×10^{-7}	1.15×10^{-6}
	Pu-238	49.8	468	1	4.0×10^{-6}	0.8	1	0.000159	0.0015
	Pu-239	4,000	756	1	4.0×10^{-6}	0.8	1	0.0128	0.00242
	Pu-240	274	25.7	1	4.0×10^{-6}	0.8	1	0.000877	0.0000822
	Pu-241	1,110	62.5	1	4.0×10^{-6}	0.8	1	0.00355	0.0002
	Am-241	90.6	0.486	1	4.0×10^{-6}	0.8	1	0.000290	1.55×10^{-6}
	Am-242M	0.940	0.000412	1	4.0×10^{-6}	0.8	1	3.01×10^{-6}	1.32×10^{-9}

Accident	Material at Risk[a]			Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies)	
	Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
Beyond-design-basis earthquake and stored fuel assembly cladding failure	H-3	2.64	62	1	1	1	1	2.64	62
	Kr-85	19.1	953	1	1	1	1	19.1	953

[a] Radionuclide Inventory from Appendix D.

Accident Scenario Descriptions for Melt and Dilute Processing—The melt and dilute process would occur in the Hot Fuel Examination Facility hot cell at ANL-W. Two melt and dilute process options are considered for ANL-W: (1) cleaning blanket spent nuclear fuel (removing metallic sodium), and (2) cleaning blanket and driver (to the extent possible) spent nuclear fuel (see Appendix C for more details). Sufficient steel would be added to both process options to form an alloy with a composition of 50 percent each of uranium and steel. Both options would occur at a temperature range of about 1,400 °C (2,550 °F). For analysis purposes, it was assumed that, on average, 120 batches of melt and dilute processing could be performed per year, considering an 80 percent availability and a three-batches-per-week operation. Each batch would process about 60 kilograms (132 pounds) of heavy metal of blanket spent nuclear fuel or about 16 kilograms (35 pounds) of driver spent nuclear fuel (diluted with depleted uranium to a 60-kilogram-equivalent [132-pound-equivalent] heavy metal). This would lead to eight years of operations for processing blanket spent nuclear fuel and two years of processing for driver spent nuclear fuel. Prior to the melt and dilute process, the sodium-bonded spent fuel elements would be cut into segments. The segmented fuel elements would be heated to a temperature above the 200 °C (392 °F) melting point of sodium and the molten sodium would be drained into a collection tank. The temperature of this bulk sodium would be raised to 690 °C (1,274 °F), to volatilize the cesium and separate it from the sodium (see Appendix C for a more detailed description of this process).

Table F–9 identifies a list of accident scenarios that were considered to be applicable to the melt and dilute process as proposed at ANL-W. These scenarios are based on the analysis of the melt and dilute process provided in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). The accident scenarios and the corresponding source terms have been modified to reflect the specifics associated with the design of the Hot Fuel Examination Facility, the characteristics of the fuel type being processed, the material at risk, and the related release fractions.

Table F–9 Selected Accident Scenarios for Melt and Dilute Processing at ANL-W

Scenario	Frequency (per year)
Nuclear criticality	0.0003
Cask drop	0.01
Waste handling accident	0.0024
Sodium fire [a]	0.008
Aircraft crash	6×10^{-7} to 1×10^{-8}
Design-basis earthquake	0.008

[a] This event is evaluated as being a direct consequence of the design-basis earthquake.

Each accident scenario description sets the condition of the accident and provides a summary of the material involved. The following paragraphs provide a summary of the accidents analyzed, the material at risk, and the release factors for the EBR-II blanket and driver spent nuclear fuel (the Fermi-1 blanket spent nuclear fuel has a very low radioactive inventory).

- *Nuclear criticality*—A criticality accident could result from the processing of multiple batches (double batching) of fissile material in the melter. This accident was considered for the driver spent nuclear fuel only. The criticality was assumed to consist of 5×10^{17} fissions (a solid criticality fission yield) (DOE 2000). The Hot Fuel Examination Facility structure would not be compromised and its ventilation system would be expected to continue to function after a criticality event. Procedural controls would be used to prevent such an accident. Therefore, such an accident would be the result of a combination of human errors, as all criticality controls are designed to meet double contingency requirements. The Hot Fuel Examination Facility Safety Analysis Report identifies a criticality event as an incredible event with an assigned frequency of less than 1×10^{-6} per year (ANL 1998b). However, this Safety Analysis Report does not specifically address melt and dilute operations. A criticality event for the SRS melt and dilute process has been addressed (DOE 2000) and, for consistency among alternatives, this analysis has been adapted. Based on the assumption of approximately 120 batches of melt and dilute operations per year and a similar frequency analysis for this type of accident at SRS, the expected frequency of this event was estimated to be 0.0003 per year for the melt and dilute operations at ANL-W. The material at risk and release fractions are provided in **Table F–10**. The damage ratio and leak path factor for the volatile, gaseous fission products were assumed conservatively to be 1. A respirable fraction value of 1 also was used. The airborne release fraction values range from 0.5 to 0.05 (DOE 1994b).

Table F–10 Melt and Dilute Process Material at Risk and Release Fraction Values for a Nuclear Criticality Event at ANL-W

Material at Risk		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Isotope	*Curies*					
Br-83	4.90	1	0.05	1	1	0.25
Br-84	16.3	1	0.05	1	1	0.82
Kr-83M	1.50	1	0.5	1	1	0.75
Kr-85M	7.2	1	0.5	1	1	3.6
Kr-87	32.8	1	0.5	1	1	17
Kr-88	32.9	1	0.5	1	1	17
Kr-89	1820	1	0.5	1	1	910
Te-129	2.70	1	0.07	1	1	0.19
Te-131	57.5	1	0.07	1	1	4.0
Te-131M	0.320	1	0.07	1	1	0.022
Te-132	1.60	1	0.07	1	1	0.11
Te-133	25.7	1	0.07	1	1	1.8
Te-133M	30.3	1	0.07	1	1	2.1
Te-134	90.5	1	0.07	1	1	6.3
I-131	0.212	1	0.05	1	1	0.011
I-132	0.855	1	0.05	1	1	0.043
I-133	6.80	1	0.05	1	1	0.34
I-134	98.0	1	0.05	1	1	4.9
I-135	22.1	1	0.05	1	1	1.1
Xe-133	0.026	1	0.5	1	1	0.013
Xe-135	2.61	1	0.5	1	1	1.3
Xe-135M	23.9	1	0.5	1	1	0.12
Xe-137	1940	1	0.5	1	1	0.097
Xe-138	665	1	0.5	1	1	0.033

- *Cask drop*—This event is similar to the cask drop event analyzed for the electrometallurgical treatment process. Spent nuclear fuel casks would be handled frequently when the sodium-bonded fuel is treated using the melt and dilute process. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) The accident involves a dropped cask during loading or unloading, seal failure, and spent nuclear fuel cladding failure sufficient to release gaseous and volatile fission products to the atmosphere, and is the same as previously described for the cask drop accident for the electrometallurgical treatment process. The material at risk and release fraction values are provided in Table F–4. (See the accident description for more detail.)

- *Waste handling accident*—The filters used in the melt and dilute off-gas exhaust system must be cleaned periodically and the resultant liquid waste disposed of. Decontamination of the filters was assumed to be performed after 10 batches are processed. Therefore, it was assumed that after processing 600 kilograms (1,320 pounds) of heavy metal of blanket spent nuclear fuel or 160 kilograms (352 pounds) of heavy metal of driver spent nuclear fuel, the filters would be decontaminated. It was postulated that a spill would occur during the transfer of the decontaminated liquid from one container to another. The event frequency is estimated at 0.0024 events per year (WSRC 1998a). The material at risk is from the fission products released during the melting process and collected on the filters. This includes fission products with boiling points at or below 1400 °C (2,550 °F) and some metal oxides that can be expected to form during the heating process (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5, respectively, were chosen for the material based on the release of material from aqueous spills (DOE 1994b). The spill was assumed to occur in an area not provided with a filtration system and, therefore, the leak path factor is 1. The material at risk, release fractions, and curies released for this accident for both EBR-II blanket and driver spent nuclear fuel are presented in **Table F–11**.

Table F–11 Melt and Dilute Process Material at Risk and Release Fraction Values for a Waste Handling Accident at ANL-W

Material At Risk [a]			*Damage Ratio* [b]	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Sr-90	484.2	31520	0.5	0.0002	0.5	1	0.024	1.58
Sb-125	13.86	473.6	0.5	0.0002	0.5	1	0.00069	0.024
Te-125M	5.71	196.8	0.5	0.0002	0.5	1	0.00029	0.0098
I-129	0.00086	0.012	0.5	0.0002	0.5	1	4.32×10^{-8}	6.0×10^{-7}
Cs-134	8.04	281.6	0.5	0.0002	0.5	1	0.000402	0.014
Cs-137	1038	35360	0.5	0.0002	0.5	1	0.0519	1.77
Pu-238	5.63	26.6	0.000015	0.0002	0.5	1	8.4×10^{-9}	4.0×10^{-8}
Pu-239	451.8	43.0	0.000015	0.0002	0.5	1	6.8×10^{-7}	6.5×10^{-8}
Pu-240	31.08	1.5	0.000015	0.0002	0.5	1	4.7×10^{-8}	2.3×10^{-9}
Pu-241	126.0	3.6	0.000015	0.0002	0.5	1	1.9×10^{-7}	5.4×10^{-9}
Am-241	9.78	0.063	0.000015	0.0002	0.5	1	1.5×10^{-8}	9.5×10^{-11}
Am-242M	0.10	0.000016	0.000015	0.0002	0.5	1	1.5×10^{-10}	2.4×10^{-14}

[a] Radionuclide inventory from Appendix D.
[b] Damage ratio values for particulates that would not be condensed in the off-gas system include a factor of 0.00003 to account for the fraction oxidized and released from liquid metals and captured on the filters.

- *Aircraft crash*—The potential for an aircraft crash was evaluated for the Hot Fuel Examination Facility and the Fuel Conditioning Facility as part of the evaluation of the electrometallurgical treatment process. The discussion provided previously is applicable to the use of the Hot Fuel Examination Facility in the melt and dilute process (see the discussion for the electrometallurgical treatment process earlier in this section). It was concluded that the likelihood of an aircraft crash causing damage to the facility process is not reasonably foreseeable; therefore, no specific analysis is needed.

- *Sodium Fire*—The sodium fire event selected for analysis was postulated to occur during the fuel cleaning (sodium removal) process for the sodium-bonded spent nuclear fuel. The event is the result of a breach in the Hot Fuel Examination Facility cell followed by a sodium fire. This event can occur as a result of the design-basis earthquake, which results in Main Cell breaches at piping and ventilation penetrations and results in a failure of the ventilation system. The frequency of this event is 0.008 per year (or once in 125 years).

  It has been estimated that approximately 10 percent of the cesium in the spent nuclear fuel has migrated from the fuel region and bonded with the sodium being removed in the fuel cleaning process. Using the radionuclide inventories provided in Appendix D for the EBR-II driver and radial blanket spent nuclear fuel and the Fermi-1 blanket spent nuclear fuel, it was estimated that a total of 670 curies of cesium-134 and 76,000 curies of cesium-137 would be entrained within the sodium. Assuming that as much as one-half of the sodium is accumulated within the collection tank prior to processing to remove cesium from the sodium, the material at risk for the sodium fire would be 340 curies of cesium-134 and 38,000 curies of cesium-137. The release fractions selected for this accident are a damage ratio of 1, an airborne release fraction and a respirable fraction each of 0.00025, and a leak path factor of 0.125. The airborne release fraction and respirable fraction value is the same as that used for cesium release from a metal fire in the design-basis seismic event analysis. The leak path factor is the value used for the Hot Fuel Examination Facility during a design-basis seismic event. The total quantity of cesium released (source term) as a result of this accident is 0.011 curies of cesium-134 and 1.2 curies of cesium-137. The cesium source term from sodium in driver fuel is 0.0095 curies of cesium-134 and 1.14 curies of cesium-137.

- *Design-basis earthquake*—This is the same accident that was developed for the design-basis earthquake for the electrometallurgical treatment process at the Hot Fuel Examination Facility. The equipment availability and damage were assumed to be the same when the facility is used for the melt and dilute process as when it is used for the electrometallurgical treatment process. Consistent with the facility safety analysis report, the ventilation system was assumed to have failed, creating a leak path factor of 0.125. The frequency of this event is 0.008 per year (or once in 125 years).

  The damage ratio, airborne release fraction, respirable fraction, and leak path factor are the same as for the electrometallurgical treatment process design-basis earthquake, with a few exceptions. Because the melt and dilute process at ANL-W operates at an elevated temperature of about 1,400 °C (2,550 °F), some fission products would boil off during the process and enter the off-gas control system. The airborne release fraction for these volatilized fission product materials, e.g., strontium, antimony, cesium, tellurium, and iodine, is set at 1 (DOE 1994b). In addition, even though some of these materials could have been condensed and removed from the off-gas system at the time of the accident, it was assumed conservatively that all of these materials would be volatilized upon initiation of the accident. However, credit is taken for the condensation of these gases as they pass through the structures on the release path. These gases will cool from their initial release temperatures as they pass through the relatively cooler structures of the Hot Fuel Examination Facility. A factor of 0.5 was used for each structure, resulting in an airborne release fraction value (representing the fraction released to the atmosphere from the cell atmosphere) of 0.125. Gaseous krypton and tritium (H-3) were not considered here, since they were assumed to have been released to the environment during the fuel cleaning process. The source terms and release fractions are provided in **Table F–12**.

Table F–12 Melt and Dilute Process Material at Risk and Release Fraction Values for a Design-Basis Earthquake at ANL-W

Material At Risk [a]			Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies)	
Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
Co-60	0.95	7.70	1	4.0×10^{-6}	0.8	0.125	3.8×10^{-7}	3.1×10^{-6}
Sr-90	48.4	3152	1	0.125	1	0.125	0.76	49
Y-90	48.4	3152	1	4.0×10^{-6}	0.8	0.125	0.000019	0.0013
Ru-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Rh-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Cd-113M	0.043	0.74	1	4.0×10^{-6}	0.8	0.125	1.7×10^{-8}	3.0×10^{-7}
Sb-125	1.39	47.36	1	0.125	1	0.125	0.022	0.74
Te-125M	0.57	19.68	1	0.125	1	0.125	0.0089	0.31
I-129	0.000086	0.0012	1	0.125	1	0.125	1.3×10^{-6}	0.000019
Cs-134	0.80	28.16	1	0.125	1	0.125	0.013	0.44
Cs-137	103.8	3536.0	1	0.125	1	0.125	1.6	55
Ba-137M	98.4	3344.0	1	4.0×10^{-6}	0.8	0.125	0.000039	0.0013
Ce-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pr-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pm-147	24.4	1321.6	1	4.0×10^{-6}	0.8	0.125	9.8×10^{-6}	0.00053
Sm-151	6.0	85.44	1	4.0×10^{-6}	0.8	0.125	2.4×10^{-6}	0.000034
Eu-154	0.44	9.07	1	4.0×10^{-6}	0.8	0.125	1.8×10^{-7}	3.6×10^{-6}
Eu-155	2.89	60.96	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	0.000024
Pu-238	0.56	2.66	1	4.0×10^{-6}	0.8	0.125	2.2×10^{-7}	1.1×10^{-6}
Pu-239	45.18	4.30	1	4.0×10^{-6}	0.8	0.125	0.000018	1.7×10^{-6}
Pu-240	3.11	0.15	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	6.0×10^{-8}
Pu-241	12.6	0.36	1	4.0×10^{-6}	0.8	0.125	5.0×10^{-6}	1.4×10^{-7}
Am-241	0.98	0.0063	1	4.0×10^{-6}	0.8	0.125	3.9×10^{-7}	2.5×10^{-9}
Am-242M	0.010	1.6×10^{-6}	1	4.0×10^{-6}	0.8	0.125	4.0×10^{-9}	6.4×10^{-13}

[a] The material at risk is the content in one batch: 60 kilograms of heavy metal of blanket fuel or 16 kilograms of heavy metal of driver fuel. Radionuclide inventory from Appendix D.

F.2.2.1.3 Accident Scenario Descriptions and Source Terms at SRS

Accident Scenario Descriptions for Plutonium-Uranium Extraction (PUREX) Processing—The following facilities would be used to store or process sodium-bonded spent nuclear fuel at SRS: F-Canyon, FB-Line, and the plutonium storage facility. The F-Canyon, FB-Line, and plutonium storage facility are part of the Building 221-F (or F-Canyon) structure. Shipments of the declad and cleaned blanket spent nuclear fuel cannot be received directly at the F-Canyon facility. The facility is not equipped to handle the transportation casks being used. The shipments would be received at the L-Reactor disassembly basin, transferred to casks suitable for shipment to F-Canyon, and then moved to F-Canyon. The PUREX process can be used to separate the plutonium from the blanket spent nuclear fuel pins. In the PUREX process, the declad and cleaned blanket spent nuclear fuel would be dissolved in the F-Canyon dissolvers, and fission products would be separated from uranium and plutonium. The plutonium solution then would be pumped to the FB-Line for purification and solidification. The depleted uranium solution would be pumped to A-Line tanks for storage and future processing into depleted uranium oxides.

The accident scenarios, identified in **Table F–13** and defined in the following paragraphs, are applicable to the processing facilities as a whole (i.e., F-Canyon and FB-Line). Transfer and storage accidents also were considered for the analysis of F-Canyon-related activities. The sodium-bonded spent nuclear fuel would be declad and cleaned prior to shipment from ANL-W. This process results in the release of gases in the gap between the fuel and cladding (see Appendix E, Section E.4), the dominant radionuclides considered during the analysis of transfer (fuel and cask drop) accidents. Therefore, the accidents were not quantified. Accidents associated with storage of the sodium-bonded spent nuclear fuel and storage of the process products (plutonium and various waste forms) were assessed as having no additional impacts beyond those associated with the process-related accidents.

Table F–13 Selected Accident Scenarios for PUREX Processing at SRS

Scenario	*Frequency (per year)*
Explosion: ion exchange column	0.0001
Nuclear criticality [a]	0.0001
Fire	0.000061
Earthquake (design-basis earthquake)	0.00013
Aircraft crash	less than 10^{-7}

[a] Only plutonium criticalities were evaluated. The potential for an americium criticality was considered but dismissed because of the limited americium mass and purity.

- *Explosion*—An explosion in an ion exchange column in the FB-Line was postulated to result from a strong exothermic reaction between nitric acid and the base resin in the cation (or anion) exchange column during plutonium solution exchange. This would result in a thermally induced pressure failure of the ion exchange vessel, and the resulting shrapnel would damage the product run tank and the product hold tank for this ion exchange pair. The explosion would breach the hot cell confinement. The plutonium in nitrite solution in the run and hold tanks would spill onto the cabinet floor and boil due to a subsequent resin fire. Based on the assumptions that the column was at its maximum load before the explosion and that the maximum quantity of liquid at the maximum allowable concentration was present, the estimated release of plutonium through the sand filter and the stack was calculated to be 0.241 grams. No other source term is applicable to the FB-Line accident. Processing in the F-Canyon would remove all other fission products before the plutonium is processed in the FB-Line (DOE 1993b). The frequency of such an event is estimated to be 0.0001 per year.

- *Fire*—In the F-Canyon Safety Analysis Report, a maximum fire was postulated to occur in the plutonium recycle process. The frequency of such a fire was estimated at 0.000061 per year (WSRC 1994). The accident was assumed to burn the contents of the largest tank. The material at risk is 86,700 kilograms (191,000 pounds) of solution. The airborne release fraction and respirable fraction were each estimated to be 0.01 (DOE 1994b). The airborne materials would pass through a sand filter, with a leak path factor of 0.005, before entering the atmosphere. The maximum recycle fire in the F-Canyon would result in the bounding source term (**Table F–14** gives the source terms). Fire in the FB-Line would result in consequences that are several times lower than those from the F-Canyon fire.

- *Nuclear criticality*—A plutonium solution criticality was postulated. The nuclear criticality was assumed to consist of an initial burst of 1×10^{18} fissions in 0.5 seconds, followed at 10-minute intervals for the next 8 hours by bursts of 2×10^{17} fissions, for a total of 1×10^{19} fissions, as specified in the U.S. Nuclear Regulatory Commission's Regulatory Guide 3.35 (NRC 1979) and NUREG-1320 (NRC 1988) and in the DOE-HDBK-3010-YR report (DOE 1994b). The 10^{19} fission yield was based on the assumptions that the solution criticality occurred in a tank with a minimum volume of 3,785 liters (1,000 gallons) and that approximately 100 liters (26 gallons) of this volume evaporated due to heat released during the fission process. Based on the data provided in the DOE Safety Survey Report (DOE 1993c), a 10^{19} criticality event in the FB-Line process would result in the bounding source term (**Table F–15** gives the source terms). The frequency of such an event was estimated to be 0.0001 per year.

Table F–14 Maximum Fire Source Terms

Isotope	*Source Term (curies)*
Sr-90	1.5
Ru-106	12
Ce-144	36
U-234	3.0×10^{-7}
U-235	4.8×10^{-6}
U-236	4.9×10^{-6}
U-238	0.00044
Pu-238	0.19
Pu-239	1.6
Pu-240	0.36
Pu-241	4.2
Pu-242	0.000053
Am-241	0.32

Source: WSRC 1994.

Table F–15 Criticality Source Terms for 10^{19} Fissions in Plutonium Solution

Isotope	*Radioactivity (curies)* [a]			*Airborne Release Fraction* [b]	*Leak Path Factor* [c]	*Source Term (curies)*
	0 to 30 Minutes	*30 Minutes to 8 Hours*	*Total*			
Kr-83m	15	95	110	1	1	110
Kr-85m	9.9	61	70.9	1	1	70.9
Kr-85	0.00012	0.00072	0.00084	1	1	0.00084
Kr-87	60	370	430	1	1	430
Kr-88	32	200	232	1	1	232
Kr-89	1,800	11,000	12,800	1	1	12,800
Xe-131m	0.014	0.086	0.1	1	1	0.1
Xe-133m	0.31	1.9	2.21	1	1	2.21
Xe-133	3.8	23	26.8	1	1	26.8
Xe-135m	460	2,800	3,260	1	1	3,260
Xe-135	57	350	407	1	1	407
Xe-137	6,900	42,000	48,900	1	1	48,900
Xe-138	1,500	9,500	11,000	1	1	11,000
I-131	1.5	9.5	11	0.25	1	2.75
I-132	170	1,000	1,170	0.25	1	293
I-133	22	140	162	0.25	1	40.5
I-134	600	3,700	4,300	0.25	1	1,075
I-135	63	390	453	0.25	1	113
Pu-238 [c, d]			3.6	0.0005	0.005	9.0×10^{-6}
Pu-239 [c, d]			170	0.0005	0.005	0.00043
Pu-240 [c, d]			39	0.0005	0.005	0.0001
Pu-241 [c, d]			2,400	0.0005	0.005	0.006
Pu-242 [c, d]			0.003	0.0005	0.005	7.50×10^{-9}

[a] Regulatory Guide 3.35 (NRC 1979).
[b] Airborne release fractions are equal to 1 for noble gases, 0.25 for iodine, and 0.0005 for plutonium; all particles were assumed to be in the respirable range (i.e., respirable fraction = 1).
[c] Plutonium in 100 liters of solution.
[d] This plutonium was assumed to be released to the atmosphere through a high-efficiency particulate air filter (e.g., SRS's sand filter) with a 0.995 efficiency. The plutonium values are the maximum solution concentration in the FB-Line (DOE 1993b).

- *Earthquake*—Recent analyses of earthquake hazards at F-Canyon indicate that a 0.24 g peak ground acceleration-level earthquake—with a return period of 8,000 years (or a frequency of 0.000125 per year) for the F-Canyon facility—could damage the structure and cause localized interior failures as well as interior and exterior wall cracks (DOE 1996b). Previous analyses of earthquake hazards at F-Canyon estimated the consequences of such an earthquake magnitude with a higher frequency of occurrences—0.0002 per year (DOE 1995b and WSRC 1994). Using the assumptions in the F-Canyon Facility Safety Analysis Report (WSRC 1994), a bounding source term was developed for an earthquake accident (**Table F–16** gives the F-Canyon source terms). Given an earthquake, it was assumed that the plutonium contents in all the processes (F-Canyon and FB-Line) would be spilled on the canyon floor. It was assumed further that the airborne material would enter the environment through the building cracks, which are formed by the loss of sealant between the sections because of differential motion of the section, with a penetration leak path factor of 0.10. For the FB-Line, the material at risk was assumed to be 2,000 grams (4.4 pounds) of plutonium in a molten metal form and 2,000 grams (4.4 pounds) of plutonium in a liquid form. The airborne release fraction multiplied by the respirable fraction is 0.0022 for the molten metal form and 0.000047 for the liquid form, including both the initial and resuspended airborne release fraction multiplied by respirable fraction values. This results in an FB-Line earthquake source term of 0.45 grams of plutonium released to the environment.

Table F–16 Maximum Earthquake Source Terms

Isotope	*Source Term (curies)*	*Isotope*	*Source Term (curies)*
Sr-90	0.086	Pu-239	0.092
Ru-106	70.1	Pu-240	0.021
Ce-144	2.05	Pu-241	0.24
Cs-137	0.0029	Pu-242	3.87×10^{-6}
Eu-154	0.017	Am-241	0.0092
Np-237	2.92×10^{-8}	Am-242m	0.000032
Np-239	0.0058	Am-243	0.0031
U-234	2.06×10^{-7}	Cm-244	0.33
U-235	2.79×10^{-7}	Cm-245	0.000027
U-236	2.81×10^{-7}	Cm-246	0.000042
U-238	0.000025	Cm-247	2.05×10^{-10}
Pu-238	0.015	—	—

Source: WSRC 1994.

- *Aircraft crash*—The F-Canyon facility is located more than 40 kilometers (25 miles) away from any major airport; therefore, no takeoff or landing crash accidents need to be considered. The crashes that could occur in flight need to be considered. According to the DOE Standard on aircraft crash analysis (DOE 1996c), the expected crash frequency for the site is approximately 0.00052 per square kilometer (0.0002 per square mile) per year from general aviation; 1.56×10^{-6} and 5.18×10^{-6} per square kilometer (6×10^{-7} and 2×10^{-6} per square mile) per year from air carrier and air taxies, respectively; and 2.59×10^{-7} and 1.56×10^{-6} per square kilometer (1×10^{-7} and 6×10^{-7} per square mile) per year from large military and small military aircraft, respectively. Using the building dimensions and the data provided in the DOE Standard for aircraft crash analysis, an upper-bound frequency for an aircraft crash into the canyon buildings was estimated to be 4.6×10^{-6} and 1.5×10^{-7} per year for general aviation and commuter (air taxi) aircraft, respectively. These values were calculated without considering any site-specific effects (e.g., the topography and building structures around the facility). Considering the available skid distance of 60 meters (200 feet) that an aircraft could skid before hitting the building, the frequency of an air taxi crashing into the building would be less than 10^{-8} per year. When only crashes that directly hit the structure were considered, general aviation aircraft would have the only estimated crash frequency greater than 10^{-7} per year. The F-Canyon building is a maximum-resistant construction structure designed to withstand a pressure of 47.9 kilopascal (1,000 pounds per square foot). Therefore, crashes of small aircraft (helicopter or a small observation/security aircraft) into these buildings are not expected to damage the buildings. If a general aviation aircraft were to crash into the buildings, its consequences (both in magnitude and frequency) would be smaller than that hypothesized for a design-basis earthquake.

Accident Scenario Descriptions for the Melt and Dilute Process—The following accidents were considered for the melt and dilute option, when performed at Building 105-L (after receipt of the declad and cleaned spent nuclear fuel at the L-Reactor Disassembly Basin), as proposed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). In this process, the declad and cleaned blanket spent nuclear fuel, along with aluminum metal, would be heated to approximately 1,000 °C (1,830 °F) to form an alloy of 30 percent uranium and 70 percent aluminum, and would be cast as ingots. The heating process would remove some of the radionuclides found in the spent nuclear fuel. The analysis assumed a batch size of 60 kilograms (132 pounds) of heavy metal, which is the batch size limit for this process when performed in Building 105-L. This would lead to three years of operations to melt and dilute the blanket fuel. The radionuclide content of an EBR-II radial blanket spent nuclear fuel batch was used conservatively to represent the radionuclide content of all blanket spent nuclear fuel. The accident scenarios identified in **Table F–17**, and described in the following paragraphs, are applicable to the melt and dilute processing of the blanket spent nuclear fuel in SRS Building 105-L. Accidents associated with the onsite transfer and storage of the declad and cleaned spent nuclear fuel were considered for analysis. As in the accident analysis for the PUREX process, these accidents were not quantified. Accidents associated with the transfer and storage of the spent nuclear fuel and diluted waste forms were assessed as having no additional impacts beyond those analyzed for process-related accidents.

Table F–17 Selected Accident Scenarios for Melt and Dilute Processing at SRS Building 105-L

Scenario	*Frequency (per year)*
Melter eruption/explosion [a]	0.0005
Waste handling spill	0.0064
Loss of electric power	0.006
Fire	0.075
Design-basis earthquake	Not applicable [b]

[a] In the draft EIS, this accident was identified as "loss of cooling water." Consistent with the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the accident name was changed.

[b] Building 105-L and the melt and dilute components are expected to remain functioning after a design-basis earthquake. The most significant impact of this event would be a potential loss of offsite power. The consequences of an earthquake up to a design-basis level thereby would be bounded by the loss-of-power event. The loss-of-power event has a higher frequency than the design-basis earthquake and is used in place of the design-basis earthquake.

- *Melter eruption/explosion*—The postulated melter eruption/explosion event could result from a buildup or addition of impurities to the metal melt. Impurities range from water (causing a steam explosion) to chemical contaminants (possible high-temperature exothermic reactions). As a result of the reaction in the metal melt, molten material would be ejected from the melter into the processing structure. Cooling water pipes within the process area could be ruptured as a result of contact with the ejected material. Should this occur, the water released would be converted to steam, resulting in an overpressurization of the enclosure that would be expected to overwhelm the exhaust fans, causing a failure of the exhaust system and an unfiltered release. Although some damage to the exhaust system is expected, there would be insufficient energy in the explosion to damage the facility structure. The melter eruption was assumed to occur with a coincident failure of the high-efficiency particulate air filtration system. The frequency of this event has been estimated to be bound by a value of 0.0005 per year (DOE 2000).

  The material at risk was estimated conservatively to be the full radionuclide content of one melt batch. The metal melt eruption/explosion was assumed to affect all the material in the melter, resulting in a damage ratio of 1 for all material. The airborne release fraction and respirable fraction values were each estimated to be 0.001 for all airborne particulates except cesium, which was estimated to be 0.2 (WSRC 2000, DOE 2000). After such an accident, the particulates would be released in the building and the ventilation fan would draw the airborne particulates to the building stack. Since the ventilation system was assumed to have failed, the leak path factor was assumed to be 1, allowing all the airborne particulates to enter the environment through the building stack. The material at risk and release fractions are summarized in **Table F–18**.

Table F–18 Melt and Dilute Process Material At Risk and Release Fraction Values for a Melter Eruption/Explosion at Building 105-L

Isotope	*Material at Risk (curies)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Sr-90	48.4	1.0	0.001	1.0	1.0	0.048
Y-90	48.4	1.0	0.001	1.0	1.0	0.048
Ru-106	8.1	1.0	0.001	1.0	1.0	0.0081
Rh-106	8.1	1.0	0.001	1.0	1.0	0.0081
Cd-113M	0.0427	1.0	0.001	1.0	1.0	0.00043
Sb-125	1.39	1.0	0.001	1.0	1.0	0.0014
Te-125M	0.571	1.0	0.081	1.0	1.0	0.0057
I-129	0.000086	1.0	1	1.0	1.0	0.000086
Cs-134	0.804	1.0	0.2	1.0	1.0	0.1608
Cs-137	104	1.0	0.2	1.0	1.0	20.8
Ba-137M	98.4	1.0	0.001	1.0	1.0	0.0984
Ce-144	3.76	1.0	0.001	1.0	1.0	0.00376
Pr-144	3.76	1.0	0.001	1.0	1.0	0.00376
Pm-147	24.4	1.0	0.001	1.0	1.0	0.0244
Sm-151	6	1.0	0.001	1.0	1.0	0.006
Eu-154	0.44	1.0	0.001	1.0	1.0	0.00044
Eu-155	2.89	1.0	0.001	1.0	1.0	0.00289
Th-228	9.30×10^{-6}	1.0	0.001	1.0	1.0	9.3×10^{-9}
U-234	0.00008	1.0	0.001	1.0	1.0	8×10^{-8}
U-235	0.000226	1.0	0.001	1.0	1.0	2.26×10^{-7}
U-236	0.000254	1.0	0.001	1.0	1.0	2.54×10^{-7}
U-238	0.0196	1.0	0.001	1.0	1.0	1.96×10^{-5}
Np-237	7.80×10^{-6}	1.0	0.001	1.0	1.0	7.80×10^{-9}
Pu-238	0.563	1.0	0.001	1.0	1.0	0.000563
Pu-239	45.2	1.0	0.001	1.0	1.0	0.0452
Pu-240	3.11	1.0	0.001	1.0	1.0	0.00311
Pu-241	12.6	1.0	0.001	1.0	1.0	0.0126
Am-241	0.978	1.0	0.001	1.0	1.0	0.000978
Am-242M	0.0101	1.0	0.001	1.0	1.0	0.000101

- *Waste handling accident*—The filters used in the melt and dilute off-gas exhaust system must be periodically cleaned and the resultant liquid waste disposed of. Decontamination of the filters was assumed to be performed after 10 batches are processed. Therefore, it was assumed that after processing 600 kilograms (1,320 pounds) of heavy metal of blanket spent nuclear fuel, the filters would be decontaminated. It was postulated that a spill would occur during the transfer of the decontaminant liquid from one container to another. The event frequency is estimated at 0.0024 per year (DOE 2000). The material at risk is from the fission products released during the melting process and collected on the filters. This includes fission products with boiling points at or below 1,000 °C (1,830 °F) and some metal oxides that can be expected to form during the heating process (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5, respectively, were chosen for the material based on the release of material from aqueous spills (DOE 1994b). The spill was assumed to occur in an area not provided with a filtration system and, therefore, the leak path factor is 1. The material at risk, release fractions, and curies released for this accident for EBR-II blanket spent nuclear fuel are presented in **Table F–19**.

Table F–19 Melt and Dilute Process Material At Risk and Release Fraction Values for a Waste Handling Accident at Building 105-L

Isotope	*Material at Risk*	*Damage Ratio* [a]	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Te-125M	5.71	0.5	0.0002	0.5	1	0.000286
I-129	0.000864	0.5	0.0002	0.5	1	4.32×10^{-8}
Cs-134	8.04	0.5	0.0002	0.5	1	0.000402
Cs-137	1040	0.5	0.0002	0.5	1	0.052
Pu-238	5.63	0.000015	0.0002	0.5	1	8.45×10^{-9}
Pu-239	452	0.000015	0.0002	0.5	1	6.78×10^{-7}
Pu-240	31.1	0.000015	0.0002	0.5	1	4.67×10^{-8}
Pu-241	126	0.000015	0.0002	0.5	1	1.89×10^{-7}
Am-241	9.78	0.000015	0.0002	0.5	1	1.47×10^{-8}
Am-242M	0.101	0.000015	0.0002	0.5	1	1.52×10^{-10}

[a] Damage ratios for neptunium, plutonium, and americium include an airborne release fraction value of 0.00003 to account for the fraction released from liquid metals and captured on the filters.

- *Loss of offsite power*—The loss of offsite power, with the subsequent failure of the onsite power supply, would result in the failure of the off-gas system, and a potential unfiltered release path to the environment. The probability of this combination of events was conservatively estimated at 0.006 per year (WSRC 1998a). The material at risk was assumed to be the volatile radionuclide inventory of one processing batch of material (approximately 60 kilograms [132 pounds] of heavy metal). Additionally, some amount of radioactive metallic and metallic oxide dusts could be generated and released during a loss-of-power event. The airborne release fraction and respirable fraction values for the gaseous fission products were each assumed to be 1, while the metallic dust release fractions at elevated temperatures are an airborne release fraction of 0.00003 and a respirable fraction of 0.04 (DOE 1994b). A leak path factor of 0.5 has been used for all material to account for possible plate-out during migration of material out of the processing area. The material at risk and release fraction values are summarized in **Table F–20**.

Table F–20 Melt and Dilute Process Material At Risk and Release Fraction Values for a Loss-of-Power Event at Building 105-L

Isotope	*Material at Risk*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Te-125M	0.571	1	1	1	0.5	0.0286
I-129	0.000086	1	1	1	0.5	0.0000432
Cs-134	0.804	1	1	1	0.5	0.0402
Cs-137	104	1	1	1	0.5	52
Pu-238	0.563	1	0.00003	0.04	0.5	3.38×10^{-7}
Pu-239	45.2	1	0.00003	0.04	0.5	0.0000271
Pu-240	3.11	1	0.00003	0.04	0.5	1.87×10^{-6}
Pu-241	12.6	1	0.00003	0.04	0.5	7.56×10^{-6}
Am-241	0.978	1	0.00003	0.04	0.5	5.87×10^{-7}
Am-242M	0.0101	1	0.00003	0.04	0.5	6.06×10^{-9}

- *Area fire*—Fires in Building 105-L have the potential to release material from the chemical decontaminate solution and the off-gas filters and baffles, and have the potential to affect the ventilation and filtration system, resulting in the release modeled for the loss-of-power event. The fire selected for analysis would result in the failure of the waste container and would release some of the decontaminate solution. This fire would have the potential to release more material than a fire that impacts the off-gas filters and baffles. The frequency of a fire in Building 105-L, based on site-wide fire data for SRS, is 0.075 fires per year. This frequency has been conservatively used as the frequency of a fire that impacts the chemical decontaminate solution. The material at risk would be the same as for the waste handling accident—the volatile gases and metallic and metallic oxide dust that would result from processing 10 batches of material in the melter. All material in the waste container would be at risk and the damage ratio was assumed to be 1. Boiling of a shallow pool of aqueous solution would result in bounding airborne release fraction and respirable fraction values of 0.002 and 1, respectively (DOE 1994b). No credit was taken for any reduction due to the leak path factor (i.e., a leak path factor of 1 was used). **Table F–21** summarizes the material at risk and release fractions for this accident scenario.

Table F–21 Melt and Dilute Process Material At Risk and Release Fraction Values for an Area Fire at Building 105-L

Isotope	*Material at Risk*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Te-125M	5.71	1	0.002	1	1	0.0114
I-129	0.00086	1	0.002	1	1	1.73×10^{-6}
Cs-134	8.04	1	0.002	1	1	0.0161
Cs-137	1040	1	0.002	1	1	2.08
Pu-238	5.63	0.00003	0.002	1	1	3.38×10^{-7}
Pu-239	452	0.00003	0.002	1	1	0.0000271
Pu-240	31.1	0.00003	0.002	1	1	1.87×10^{-6}
Pu-241	126	0.00003	0.002	1	1	7.56×10^{-6}
Am-241	9.78	0.00003	0.002	1	1	5.87×10^{-7}
Am-242M	0.101	0.00003	0.002	1	1	6.06×10^{-9}

F.2.2.2 Consequences and Risk Calculations

Once the source term for each accident scenario is determined, the radiological consequences are calculated. The calculations vary depending on how the release is dispersed, what material is involved, and which receptor is being considered. Risks are calculated based on the accident's frequency and its consequences. The risks also are stated in terms of additional cancer fatalities resulting from a release, using a conversion factor of 0.0005 latent cancer fatalities per person-rem for members of the public and 0.0004 latent cancer fatalities per person-rem for workers.

Radiological consequences to four different receptors were evaluated: a maximally exposed offsite individual (an individual member of the public), the general population, a noninvolved worker, and an involved (facility) worker. The consequences to the facility workers were qualitatively evaluated. For the other receptors, quantitative estimates of consequences were made. Two types of dispersion conditions were considered—95th percentile and 50th percentile meteorological conditions. The 50th percentile condition represents the median meteorological condition and is defined as that for which more severe conditions occur 50 percent of the time. The 95th percentile condition represents relatively low-probability meteorological conditions that produce higher calculated exposures; it is defined as that condition not exceeded more than 5 percent of the time. Both dispersion conditions were modeled using the GENII program, which determines the desired condition from the site-specific meteorological data in the form of a joint frequency distribution. Joint frequency data are usually produced from at least three consecutive years of site weather data in terms of percentage of time that the wind blows in specific directions (e.g., south, south-southwest, southwest) for the given midpoint (or average) wind speed class and atmospheric stability.

Radiological consequences to a receptor from an accident in the FB-Line were estimated based on a calculated 50-year committed dose factor (dose factor) resulting from releases of 1 gram of plutonium with an isotopic distribution associated with the EBR-II blanket spent nuclear fuel (**Table F–22**). This was done because the FB-Line processes only plutonium already separated in the F-Canyon.

The values given in this table represent the maximum dose to the receptor and were obtained using the GENII program.

Table F–22 Receptors' Dose Factors for Accidental Releases of 1 Gram of Plutonium From an Accident Initiated in the FB-Line

Receptor		*95th Percentile Meteorological Condition*	*50th Percentile Meteorological Condition*
Maximally exposed offsite individual (rem)	Elevated release	0.027	Not applicable
	Ground release	0.13	Not applicable
Population (person-rem)	Elevated release	1500	220
	Ground release	5000	270
Noninvolved worker (rem)	Elevated release	Not applicable	0.080
	Ground release	Not applicable	2

Consequences to involved workers were qualitatively assessed. This approach was used for two reasons: first, no adequate method exists for calculating meaningful consequences at or near the location where the accident could occur. Second, safety assurance for workers is demonstrated by both the workers' training and by the establishment of an Occupational Safety and Health Administration process safety management system (29 CFR 1910.119), the evaluations required by such a system, and the products derived from such evaluations (e.g., procedures, programs, emergency plans).

The consequences to the involved worker, presented in **Tables F-23** and **F–24**, are accident-dependent and site-specific. In facilities where the involved worker activities include remote operations, the consequences of accidents would be lower than in facilities where the workers are near the process. The following paragraphs summarize the various potential consequences to the involved workers from the hypothesized accidents at different sites. Additionally, a limited number of fatalities could occur in an indirect or secondary manner—for example, the involved worker could be killed by an earthquake or explosion.

Table F–23 Involved Worker Consequences From Various Hypothesized Accidents

Accident	*Consequences*
Explosion (ion exchange)	Could potentially result in fatal injuries (nonradiological) to the involved worker (SRS only).
Criticality	Could potentially result in a fatal dose to the involved worker. (Worker location outside cells, e.g., outside the argon cell at ANL-W, provides worker protection.)
Fire	No fatality is expected; some workers could inhale the dispersed radioactive materials before using a respirator and leaving the area.
Earthquake	No fatality is expected.
Spill	Involved workers could inhale the dispersed radioactive materials before using a respirator and leaving the area.

Table F–24 Involved Worker Summary

Accident Description	*Number of Workers at F-Canyon and FB-Line*	*Number of Workers at ANL-W*
SRS—PUREX Process		
Earthquake	47	50
Explosion, ion exchange column	16	Not applicable
Nuclear criticality	16	15
Fire	16	4

- *Explosion*—An explosion could result in serious, even fatal, injuries to involved workers from the accident itself. Some of the involved workers could inhale the dispersed radioactive material before using their respirators and evacuating the area. No fatality is expected from the radiological consequences.

- *Fire*—Involved workers could inhale some radioactive material before evacuating the area. No fatality is expected from the radiological consequences.

- *Spill*—Depending on the location of the spill, nearby workers could inhale the airborne radioactive materials before evacuating the area. Involved workers normally would be wearing respirators when handling the radioactive material containers. No fatality is expected to result from such an accident.

- *Earthquake*—Involved workers could receive lethal injuries from the accident itself. No fatality is expected from radiological consequences.

- *Aircraft Crash*—Consequences similar to those of an earthquake could result from the accident.

- *Criticality*—Involved workers could receive substantial, or potentially fatal, doses from prompt neutrons and gamma rays emitted from the first pulse. After the initial pulse, the workers would evacuate the area immediately on the initiation of the criticality monitoring alarms.

Analysis Conservatism and Uncertainty

To assist in evaluating the impacts of the processing options at SRS and ANL-W on a common basis, a spectrum of generic accidents was postulated for each process location. The accident scenarios were based on similar accidents documented in various site documents. When required, accident assumptions were modified to enable comparison between the sites. In cases where similar accidents were evaluated in site-specific documents, the more conservative analysis assumptions were used for all sites to normalize the results for the purpose of comparison. The following accident analysis parameters have a major impact on accident consequence estimates (i.e., the doses to workers and the public): weather conditions existing at the time of the accident, the material at risk, the isotopic breakdown of the material at risk, and the source term released to the environment.

Weather conditions assumed at the time of the accident have a large impact on dose estimates. Accident impacts to the public (both the maximally exposed offsite individual and the population) presented in this appendix were estimated using both 95^{th} percentile and median 50^{th} percentile weather conditions. The impacts presented in the body of the EIS are based on the 50^{th} percentile weather conditions for the population dose (NRC 1976) and 95^{th} percentile weather conditions (NRC 1983) for the maximally exposed offsite individual dose (which provides conservative maximally exposed offsite individual dose estimates). The GENII computer code was used to calculate doses to the public within 80 kilometers (50 miles) of the accident release point. The code calculates the public dose in each of 16 sectors centered at the accident release point. The GENII computer code also assumes that the total source term is released into each sector and that there is no change in the weather (i.e., wind direction, wind speed, and stability class) while the accident plume is traversing the 80-kilometer sector. The use of the 95^{th} percentile weather data rather than the expected or median 50^{th} percentile weather data was considered to be unrealistic for estimating the population dose. Meteorological conditions used in the analysis are based on measured weather data at the site. The 95^{th} percentile represents a very stable site meteorological condition, which cannot be expected to be applicable for a wide area up to 80 kilometers from the site. Therefore, the 50^{th} percentile, which represents a more neutral weather condition, is more representative of expected weather conditions over a wide area.

Uncertainties in accident frequencies do not impact the accident consequences, but do impact accident risk. The site/facility-specific accident frequencies (i.e., earthquake-induced building damage and aircraft crash) were based on data provided by the sites. Process-specific accident frequencies were estimated based on analyses provided in site-specific documentation. In cases where similar accidents were evaluated in site-specific documents, the more conservative accident frequency was used for all sites to normalize the results for the purpose of comparison.

Due to the layers of conservatism built into the accident analysis for the spectrum of postulated accidents, the estimated consequences and risks to the public represent the upper limit for the individual classes of accidents. The uncertainties associated with the accident frequency estimates are enveloped by the analysis conservatism.

F.2.3 Accident Analyses Consequences and Risk Results

F.2.3.1 No Action Alternative

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the interior of the fuel elements) except for stabilization activities that may be necessary for continued safe and secure storage until 2035 or until a new treatment technology is developed. Under the Electrometallurgical Treatment Research and Demonstration Project, approximately 0.4 metric tons of heavy metal of EBR-II driver spent nuclear fuel and 1.2 metric tons of heavy metal of blanket spent nuclear fuel were processed. This EIS evaluates the impacts associated with activities required to clean up and stabilize any residual waste materials generated during the demonstration project at ANL-W. In addition, at the completion of the project, any remaining sodium-bonded spent nuclear fuel in the process facilities would be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility. Spent nuclear fuel transfer activities and waste processing activities would be completed in about two years after equipment installation. Some of the spent nuclear fuel handling and processing accidents identified under Alternative 1 are applicable to the No Action Alternative. **Tables F-25** and **F-26** provide the dose calculation results for the design-basis and beyond-design-basis earthquakes for stabilizing the residual waste. The results for the remaining accidents considered for the No Action Alternative (the salt powder spill in the Hot Fuel Examination Facility, the cask drop, and the transuranic waste fire) are provided in the discussion of "Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W." Consequence and risk results are provided for the maximally exposed offsite individual, a noninvolved worker, and the general population. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver spent nuclear fuel characteristics (radionuclide compositions), which bound the consequences, were used to represent the consequences and risks during stabilization of waste for the demonstration project for the No Action Alternative. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition.

Table F–25 Summary of Dose Calculation Results for the Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Design-basis earthquake	0.008	Dose per event	12	52	0.63	0.64	4.7	1.4	0.017
		Dose per year	0.095	0.42	0.005	0.0051	0.038	0.011	0.00014
		LCF per year	4.8×10^{-8}	0.00021	2.5×10^{-9}	2.6×10^{-9}	1.5×10^{-8}	5.6×10^{-6}	6.8×10^{-8}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality

Table F–26 Summary of Dose Calculation Results for the Beyond-Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Beyond-design-basis earthquake [a]	0.00001	Dose per event	96	42	5.1	5.1	37	11	0.13
		Dose per year	0.00096	0.00042	0.000051	0.000051	0.00037	0.00011	1.3×10^{-6}
		LCF per year	4.8×10^{-10}	2.1×10^{-7}	2.6×10^{-11}	2.6×10^{-11}	1.5×10^{-10}	5.5×10^{-8}	6.5×10^{-13}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

[a] During stabilization of the demonstration project waste, only the Hot Fuel Examination Facility salt powder spill would be applicable.

F.2.3.2 Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W

The processing technology considered for this alternative consists solely of the electrometallurgical treatment processing of the sodium-bonded spent nuclear fuel at ANL-W, using the Fuel Conditioning Facility and the Hot Fuel Examination Facility. **Tables F–27** through **F–37** provide the dose calculation results for the electrometallurgical treatment-related accidents at ANL-W. Consequence and risk results are provided for the maximally exposed offsite individual, a noninvolved worker, and the general population. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition, rather than either a blanket or driver spent nuclear fuel-specific composition.

Table F–27 Summary of Dose Calculation Results for a Salt Powder Spill (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Hot Fuel Examination Facility salt powder spill	0.01	Dose per event	0.00046	0.0026	0.000031	0.000046	4.7×10^{-7}	0.000098	1.2×10^{-6}
		Dose per year	4.6×10^{-6}	0.000026	3.1×10^{-7}	4.6×10^{-7}	4.7×10^{-9}	9.8×10^{-7}	1.2×10^{-8}
		LCF per year	2.3×10^{-12}	1.3×10^{-8}	1.6×10^{-13}	2.3×10^{-13}	1.9×10^{-15}	4.9×10^{-10}	5.9×10^{-15}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–28 Summary of Dose Calculation Results for a Salt Powder Spill (Blanket)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Hot Fuel Examination Facility salt powder spill	0.01	Dose per event	0.00015	0.00088	0.000011	0.000015	1.3×10^{-6}	0.000033	4.0×10^{-7}
		Dose per year	1.5×10^{-6}	8.8×10^{-6}	1.1×10^{-7}	1.5×10^{-7}	1.3×10^{-8}	3.3×10^{-7}	4.0×10^{-9}
		LCF per year	7.5×10^{-13}	4.4×10^{-9}	5.5×10^{-14}	7.5×10^{-14}	5.3×10^{-15}	1.7×10^{-10}	2.0×10^{-15}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–29 Summary of Dose Calculation Results for a Cask Drop (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Cask drop	0.01	Dose per event	0.03	0.14	0.0017	0.0016	0.00084	0.0035	0.000042
		Dose per year	0.0003	0.0014	0.000017	0.000016	8.4×10^{-6}	0.000035	4.2×10^{-7}
		LCF per year	1.5×10^{-10}	7.0×10^{-7}	8.5×10^{-12}	8.0×10^{-12}	3.4×10^{-12}	1.7×10^{-8}	2.1×10^{-13}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–30 Summary of Dose Calculation Results for a Cask Drop (Blanket)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Cask drop	0.01	Dose per event	0.0024	0.011	0.00013	0.00013	0.000049	0.00028	3.4×10^{-6}
		Dose per year	0.000024	0.00011	1.3×10^{-6}	1.3×10^{-6}	4.9×10^{-7}	2.8×10^{-6}	3.4×10^{-8}
		LCF per year	1.2×10^{-11}	5.5×10^{-8}	6.5×10^{-13}	6.5×10^{-13}	2.0×10^{-13}	1.4×10^{-9}	1.7×10^{-14}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–31 Summary of Dose Calculation Results for a Single-Container Transuranic Waste Fire

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Transuranic waste fire	0.001	Dose per event	0.059	0.27	0.0033	0.0032	0.22	0.0071	0.000085
		Dose per year	0.000059	0.00027	3.3×10^{-6}	3.2×10^{-6}	0.00022	7.1×10^{-6}	8.5×10^{-8}
		LCF per year	3.0×10^{-11}	1.4×10^{-7}	1.6×10^{-12}	1.6×10^{-12}	8.8×10^{-11}	3.6×10^{-9}	4.3×10^{-14}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–32 Summary of Dose Calculation Results for a Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Design-basis earthquake	0.0002 (Multi-facility event)	Dose per event	13	70	0.84	0.95	4.7	2.8	0.034
		Dose per year	0.0026	0.014	0.00017	0.00019	0.00084	0.00056	6.8×10^{-6}
		LCF per year	1.3×10^{-9}	7.0×10^{-6}	8.4×10^{-11}	9.5×10^{-11}	3.8×10^{-10}	2.8×10^{-7}	3.4×10^{-12}
	0.008 (HFEF)	Dose per event	12	52	0.63	0.64	4.7	1.4	0.017
		Dose per year	0.095	0.42	0.0050	0.0051	0.037	0.011	0.00013
		LCF per year	4.8×10^{-8}	0.00021	2.5×10^{-9}	2.6×10^{-9}	1.5×10^{-8}	5.6×10^{-6}	6.6×10^{-11}

MEI = Maximally Exposed Offsite Individual, HFEF = Hot Fuel Examination Facility, LCF = Latent Cancer Fatality.

Table F–33 Summary of Dose Calculation Results for a Design-Basis Earthquake (Blanket)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Design-basis earthquake	0.0002 (Multi-facility event)	Dose per event	4.1	18	0.22	0.23	14	0.49	0.006
		Dose per year	0.00081	0.0036	0.000044	0.000046	0.0027	0.00010	1.2×10^{-6}
		LCF per year	4.1×10^{-10}	1.8×10^{-6}	2.2×10^{-11}	2.3×10^{-11}	1.1×10^{-9}	4.9×10^{-8}	6.0×10^{-13}
	0.008 (HFEF)	Dose per event	4.0	18	0.21	0.22	14	0.47	0.0057
		Dose per year	0.032	0.14	0.0017	0.0018	0.11	0.0038	0.000045
		LCF per year	1.6×10^{-8}	0.000072	8.6×10^{-10}	8.8×10^{-10}	4.5×10^{-8}	1.9×10^{-6}	2.3×10^{-11}

MEI = Maximally Exposed Offsite Individual, HFEF = Hot Fuel Examination Facility, LCF = Latent Cancer Fatality.

Table F–34 Summary of Dose Calculation Results for a Salt Transfer Drop (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Salt transfer drop	1.0×10^{-7}	Dose per event	0.19	0.84	0.01	0.01	0.073	0.022	0.00026
		Dose per year	1.9×10^{-8}	8.4×10^{-8}	1.0×10^{-9}	1.0×10^{-9}	7.3×10^{-9}	2.2×10^{-9}	2.6×10^{-11}
		LCF per year	9.5×10^{-15}	4.2×10^{-11}	5.0×10^{-16}	5.0×10^{-16}	2.9×10^{-15}	1.1×10^{-12}	1.3×10^{-17}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–35 Summary of Dose Calculation Results for a Salt Transfer Drop (Blanket)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Salt transfer drop	1.0×10^{-7}	Dose per event	0.065	0.29	0.0035	0.0036	0.22	0.0077	0.000092
		Dose per year	6.5×10^{-9}	2.9×10^{-8}	3.5×10^{-10}	3.6×10^{-10}	2.2×10^{-8}	7.7×10^{-10}	9.2×10^{-12}
		LCF per year	3.3×10^{-15}	1.5×10^{-11}	1.8×10^{-16}	1.8×10^{-16}	8.8×10^{-15}	3.9×10^{-13}	4.6×10^{-18}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–36 Summary of Dose Calculation Results for a Beyond-Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Beyond-design-basis earthquake	0.00001	Dose per event	22,000	97,000	1,200	1,200	370	2,500	31
		Dose per year	0.22	0.97	0.012	0.012	0.0037	0.025	0.00031
		LCF per year	2.2×10^{-7}	0.00049	5.9×10^{-9}	6.0×10^{-9}	1.5×10^{-9}	0.000013	1.5×10^{-10}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–37 Summary of Dose Calculation Results for a Beyond-Design-Basis Earthquake (Blanket)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Beyond-design-basis earthquake	0.00001	Dose per event	930	4,200	51	50	560	110	1.3
		Dose per year	0.0093	0.042	0.00051	0.00050	0.0056	0.0011	0.000013
		LCF per year	4.7×10^{-9}	0.000021	2.5×10^{-10}	2.5×10^{-10}	2.3×10^{-9}	5.5×10^{-7}	6.5×10^{-12}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.2.3.3 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

The processing technology considered for this alternative consists of cleaning the sodium from blanket spent nuclear fuel and packaging the cleaned blanket spent nuclear fuel in high-integrity cans. The sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process. The dose calculation results for this combination of processes at ANL-W are found in Section F.2.3.2 for driver spent nuclear fuel and in F.2.3.4 for blanket spent nuclear fuel. All of the electrometallurgical treatment accidents for the driver spent nuclear fuel are applicable to this process. For the blanket spent nuclear fuel, the sodium fire and the cask handling accident are applicable. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver spent nuclear fuel and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

F.2.3.4 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

The processing technology considered for this alternative consists of decladding and cleaning the sodium-bonded blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W and shipment of this material to SRS for PUREX processing. In this alternative, the sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process at ANL-W. **Tables F–38** through **F–44** provide the dose calculation results for accidents during PUREX processing at SRS and for cask drop and sodium fire accidents at ANL-W. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

Consequence and risk estimates are provided for both processing the blanket spent nuclear fuel material at ANL-W prior to its shipment to SRS and for processing the material at SRS. Analysis results for processing the driver spent nuclear fuel can be found in the discussion for Alternative 1 in Section F.2.3.2.

Table F–38 Summary of Dose Calculation Results for an F-Canyon Fire

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
F-Canyon fire	0.000061	Dose per event	610	36,000	2,300	5,500
		Dose per year	0.037	2.2	0.14	0.34
		LCF per year	1.9×10^{-8}	0.0011	5.6×10^{-8}	0.00017

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–39 Summary of Dose Calculation Results for an FB-Line Explosion

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
FB-Line explosion	0.00010	Dose per event	6.5	360	19	53
		Dose per year	0.00065	0.036	0.0019	0.0053
		LCF per year	3.3×10^{-10}	0.000018	7.6×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–40 Summary of Dose Calculation Results for an F-Canyon Earthquake

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
F-Canyon earthquake	0.00013	Dose per event	1,100	38,000	12,000	2,100
		Dose per year	0.14	4.9	1.56	0.27
		LCF per year	7.2×10^{-8}	0.0025	6.2×10^{-7}	0.00014

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–41 Summary of Dose Calculation Results for an FB-Line Earthquake

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
FB-Line earthquake	0.00013	Dose per event	58	2,250	900	120
		Dose per year	0.0075	0.29	0.12	0.016
		LCF per year	3.8×10^{-9}	0.00015	4.7×10^{-8}	7.8×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–42 Summary of Dose Calculation Results for an F-Canyon Criticality Accident

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
F-Canyon criticality	0.00010	Dose per event	11	380	37	59
		Dose per year	0.0011	0.038	0.0037	0.0059
		LCF per year	5.5×10^{-10}	0.000019	1.5×10^{-9}	3.0×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–43 Summary of Dose Calculation Results for an ANL-W Cask Drop Accident

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Cask drop	0.01	Dose per event	0.0024	0.011	0.000049	0.00028
		Dose per year	0.000024	0.00011	4.9×10^{-7}	2.8×10^{-6}
		LCF per year	1.2×10^{-11}	5.5×10^{-8}	2.0×10^{-13}	1.4×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–44 Summary of Dose Calculation Results for an ANL-W Sodium Fire

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Sodium fire during decladding and cleaning	0.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF per year	2.4×10^{-8}	0.00011	1.7×10^{-10}	2.8×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.2.3.5 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

The processing technology considered for this alternative consists of melting and diluting the cleaned blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W. In this alternative, the sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process at ANL-W. The dose calculation results for this alternative are provided in this section. The results for the driver spent nuclear fuel are presented as part of the results for Alternative 1 (Section F.2.3.2) and the results for the blanket spent nuclear fuel are presented as part of the results for Alternative 6 (Section F.2.3.7), where the results for melt and dilute processing of both driver and blanket spent nuclear fuel are presented. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

F.2.3.6 Alternative 5: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS

The processing technology considered for this alternative consists of decladding, cleaning, and packaging the blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W and shipping the packaged blanket spent nuclear fuel to SRS for melt and dilute processing in Building 105-L. In this alternative, the sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process at ANL-W. **Tables F–45** through **F–50** provide the dose calculation results for the melt and dilute process at SRS. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

Consequence and risk estimates are provided for both processing the blanket spent nuclear material at ANL-W prior to its shipment to SRS, and for processing the material at SRS. Analysis results for processing driver spent nuclear fuel can be found in the discussion for Alternative 1 in Section F.2.3.2.

Table F–45 Summary of Dose Calculation Results for an L-Area Waste Handling Accident

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Waste handling spill	0.0064	Dose per event	2.1	42	0.17	3.6
		Dose per year	0.013	0.27	0.0011	0.023
		LCF per year	6.7×10^{-9}	0.000014	5.5×10^{-10}	0.000012

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–46 Summary of Dose Calculation Results for a Building 105-L Loss-of-Power Event

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Loss-of-power event	0.006	Dose per event	2,100	42,000	140	3,500
		Dose per year	12.6	250	0.84	21
		LCF per year	6.3×10^{-6}	0.13	3.4×10^{-7}	0.011

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–47 Summary of Dose Calculation Results for a Building 105-L Melter Eruption/Explosion

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Melter eruption/ explosion	0.0005	Dose per event	269	6,390	72.9	1,160
		Dose per year	0.14	3.2	0.037	0.58
		LCF per year	7.0×10^{-8}	0.0016	1.5×10^{-8}	0.00029

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–48 Summary of Dose Calculation Results for a Building 105-L Fire

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Fire	0.075	Dose per event	86	1,700	6.3	140
		Dose per year	6.5	130	0.47	11
		LCF per year	3.2×10^{-6}	0.064	1.9×10^{-7}	0.0053

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–49 Summary of Dose Calculation Results for an ANL-W Cask Drop Accident

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Cask drop	0.01	Dose per event	0.0024	0.011	0.000049	0.00028
		Dose per year	0.000024	0.00011	4.9×10^{-7}	2.8×10^{-6}
		LCF per year	1.2×10^{-11}	5.5×10^{-8}	2.0×10^{-13}	1.4×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–50 Summary of Dose Calculation Results for an ANL-W Sodium Fire

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
Sodium fire during decladding and cleaning	0.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF per year	2.4×10^{-8}	0.00011	1.7×10^{-10}	2.8×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.2.3.7 Alternative 6: Melt and Dilute Blanket and Driver Fuel at ANL-W

The processing technology considered for this alternative consists of cleaning both blanket and driver spent nuclear fuel and melting and diluting the spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W. **Tables F–51** through **F–57** provide the dose calculation results for the melt and dilute process at ANL-W. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

Consequence and risk estimates are provided for both the declad and clean processing and the melt and dilute processing of the sodium-bonded spent nuclear fuel.

Table F–51 Summary of Dose Calculation Results for a Melt and Dilute Design-Basis Event (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Design-basis earthquake (includes sodium fire)	0.008	Dose per event	19,000	89,400	1,080	1,080	838	2,250	27
		Dose per year	152	715.2	8.64	8.64	6.7	18	0.22
		LCF per year	0.000076	0.36	4.3×10^{-6}	4.3×10^{-6}	2.7×10^{-6}	0.0090	1.1×10^{-7}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–52 Summary of Dose Calculation Results for a Melt and Dilute Design-Basis Event (Blanket)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Design-basis earthquake (includes sodium fire)	0.008	Dose per event	471	2240	26.9	27	15.2	56.1	0.68
		Dose per year	3.8	17.92	0.22	0.22	0.12	0.45	0.0054
		LCF per year	1.9×10^{-6}	0.0090	1.1×10^{-7}	1.1×10^{-7}	4.8×10^{-8}	0.00022	2.7×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–53 Summary of Dose Calculation Results for a Melt and Dilute Waste Handling Accident (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***	***MEI (millirem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Waste handling accident (liquid spill)	0.0024	Dose per event	597	2820	34	33.9	26.7	70.8	.85
		Dose per year	1.43	6.77	0.082	0.081	0.064	0.17	0.0020
		LCF per year	7.2×10^{-7}	0.0034	4.1×10^{-8}	4.1×10^{-8}	2.6×10^{-8}	0.000085	1.0×10^{-9}

MEI - Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–54 Summary of Dose Calculation Results for a Melt and Dilute Waste Handling Accident (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Waste handling accident (liquid spill)	0.0024	Dose per event	14.9	70.8	0.85	0.85	0.49	1.8	0.022
		Dose per year	0.036	0.17	0.0020	0.0020	0.0012	0.0043	0.000053
		LCF per year	1.8×10^{-8}	0.000085	1.0×10^{-9}	1.0×10^{-9}	4.8×10^{-10}	2.2×10^{-6}	2.7×10^{-11}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–55 Summary of Dose Calculation Results for a Melt and Dilute Criticality Accident (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Criticality	0.0003	Dose per event	0.52	1.6	0.019	0.083	0.47	0.085	1.0×10^{-6}
		Dose per year	0.00016	0.00048	0.0000057	0.000025	0.00014	0.000026	3.0×10^{-10}
		LCF per year	8.0×10^{-11}	2.4×10^{-7}	2.9×10^{-12}	1.3×10^{-11}	5.6×10^{-11}	1.3×10^{-8}	1.5×10^{-16}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–56 Summary of Dose Calculation Results for a Melt and Dilute Sodium Fire (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Sodium fire	0.008	Dose per event	282	1,260	15.2	15.6	2.59	33	0.4
		Dose per year	0.23	10.08	0.12	0.12	0.021	0.26	0.0032
		LCF per year	1.13×10^{-6}	0.0050	6.0×10^{-8}	6.0×10^{-8}	8.3×10^{-9}	0.00013	1.6×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–57 Summary of Dose Calculation Results for a Melt and Dilute Sodium Fire (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Sodium fire	0.008	Dose per event	5.9	26.3	0.32	0.33	0.054	0.69	0.0083
		Dose per year	0.047	0.21	0.0026	0.0026	0.00043	0.0055	0.000066
		LCF per year	2.4×10^{-8}	0.00011	1.3×10^{-9}	1.3×10^{-9}	1.7×10^{-10}	2.8×10^{-6}	3.3×10^{-11}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.3 IMPACTS OF HAZARDOUS CHEMICAL ACCIDENTS ON HUMAN HEALTH

F.3.1 Chemical Accident Analysis Methodology

Factors such as receptor location, terrain, meteorological conditions, release conditions, and characteristics of the chemical inventory are required as input parameters for hand calculations or computer codes to determine human exposure from airborne releases of toxic chemicals. This section gives a general narrative about these input parameters with degrees of conservatism noted, and describes the computer models used to perform exposure estimates. EPlcode™ is the computer code chosen for estimating airborne concentrations resulting from most releases of toxic chemicals (Homann 1988).

F.3.1.1 *EPIcode™*

EPIcode™ uses the well-established Gaussian Plume Model to calculate the airborne toxic chemical concentrations at the receptor locations. The EPIcode™ library contains information on over 600 toxic substances listed in the *Threshold Limit Values for Chemical Substances and Physical Agents and Biomedical Exposure Indices* (ACGIH 1994). The types of releases that can be modeled, and associated input parameters, are discussed below.

Continuous release models require specifying the source term as an ambient concentration and a release rate. For term releases, the user specifies the release duration and the total quantity of material released. Area continuous and area term releases are useful in calculating the effects of a release from pools of spilled volatile liquids. The user must enter the effective radius of the release (i.e., the radius of the circle encompassing the spill area). Also entered is the temperature of the pool and ambient temperature to establish the release rate from a liquid spill. An upwind virtual point source, which results in an initial lateral diffusion equal to the effective radius of the area source, is used to model an area release.

By specifying a release quantity, duration, and area, the user effectively proposes a release rate per unit spill area. EPIcode™ confirms that the volatility of the spilled substance can support such a release rate. If the proposed release rate exceeds the saturation conditions at the release temperature, the EPIcode™ calculates a lower release rate and a corresponding longer release time.

In calculating effective release height, the actual plume height may not be the physical release height (e.g., the stack height). Plume rise can occur because of the velocity of a stack emission and the temperature differential between the stack effluent and the surrounding air. EPIcode™ calculates both the momentum and buoyant plume rise and chooses the greater of the two results.

Concentrations of chemical and radiological materials are highly dependent upon the effective release height (e.g., the effective height of a stack or an evaporating pool of spilled material). Thermal buoyancy was taken into consideration for those scenarios involving fire or heat sources. In those cases, a temperature of 200 °C (392 °F) was assumed for the thermal buoyancy term. This is conservative, since expected surface temperatures and resulting buoyancy terms are expected to be greater in actual fires or heat sources.

In this application, the standard terrain calculation of EPIcode™ is always used. Except as otherwise noted, both the 50^{th} and 95^{th} percentile meteorological (stability class and wind speed) conditions for INEEL were input into EPIcode™. The receptor height is always ground level (0 meters) and the mixing layer height is always 400 meters (1,300 feet).

As described in its user manual (Homann 1988), the EPIcode™ also performs the following steps:

- Treats a release as instantaneous versus continuous, depending upon the plume length at the specific downwind location being considered
- Corrects the concentration for sampling time
- Adjusts the wind speed for release height

- Depletes the plume as a function of downwind distance
- Adjusts the standard deviations of the crosswind and vertical concentrations for brief releases

As output, EPIcode™ can generate data plots of mean toxic chemical concentration (during a specified averaging time) as a function of downwind distance. From these graphs and numerical output, the concentrations at receptor locations are determined and evaluated for health effects.

EPIcode™ was selected as the computer code for release analysis of chemicals amenable to Gaussian modeling after comparison with a number of codes, primarily CHARM and ARCHIE. It was judged easier to use for this simple application than either the more sophisticated, proprietary CHARM code or the comparable, public domain ARCHIE code. The SLAB code had previously been selected by INEEL as the most appropriate of the refined dispersion models (such as CHARM) for modeling special case releases, such as dense gas dispersion, where negative buoyancy effects must be considered. However, because chemical accident scenarios involving dispersion of denser-than-air gases were not considered in this analysis, the SLAB model was not used. EPlcode™ was judged to be a satisfactory code for the inventory of chemicals analyzed.

F.3.1.2 Health Effects

Hazardous constituents dispersed during an accident could induce adverse health effects among exposed individuals. This possible impact is assessed by comparing the airborne concentrations of each substance at specified downwind receptor locations to standard accident exposure guidelines for chemical toxicity.

Where available, the Emergency Response Planning Guideline (ERPG) values were used for this comparison. The guideline values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects. The ERPG values are specific for each substance, and are derived for each of three general severity levels:

- *ERPG-1:* The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

- *ERPG-2:* The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

- *ERPG-3:* The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

Where ERPG values were not derived for a toxic substance, other chemical toxicity values were substituted, as follows:

- For ERPG-1, threshold-limit value/time-weighted average values (ACGIH 1994) were substituted: The time-weighted average is the concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect.

- For ERPG-2, level-of-concern values (equal to 0.1 of immediately-dangerous-to-life-or-health values) were substituted: "level of concern" is defined as the concentration of a hazardous substance in air, above which there could be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (EPA 1987).

- For ERPG-3, immediately-dangerous-to-life-or-health values were substituted: "immediately dangerous to life or health" is defined as the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any escape-impairing or irreversible side effects (HHS 1997).

Possible health effects associated with exceeding an ERPG-2 or -3 value are specific for each substance of concern, and must be characterized in that context. When concentrations are found to exceed an ERPG or substitute value, specific toxicological effects for the chemicals of concern are considered in describing possible health effects associated with exceeding a threshold value.

The ERPG values are based upon a one-hour exposure of a member of the general population. In this analysis, the ERPG values were applied only to time-averaged exposures of one hour or less in duration. This approach provides an additional element of conservatism in the evaluation of accidents with releases that are significantly less than one hour. In instances of very short exposures to substances whose effects are concentration-dependent (e.g., chlorine) and where toxicological data support analysis at short exposure times, threshold concentrations of lethality are reported (the minimum concentration necessary to cause a fatality).

F.3.2 Accident Scenario Selection and Descriptions

F.3.2.1 Toxic Chemical Accidents at ANL-W

This section describes the nonradiological consequences of the abnormal event associated with handling uranium ingots. Four accidents have been identified at ANL-W that have the potential to result in the release of either uranium or uranium and cadmium. These accidents, a uranium handling accident, a design-basis uranium fire, a design-basis earthquake, and a beyond-design-basis earthquake, are discussed below.

F.3.2.1.1 Uranium Handling Accident

Uranium ingots (20 percent enrichment or less) from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to onsite storage at the Zero Power Physics Reactor Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms (310 pounds) of ingots. Depleted uranium also is stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a handling accident in which an ingot of uranium is dropped onto a hard surface, small particles are broken off the ingot, and the pyrophoric properties of the uranium result in ignition of the particles. The resulting small fire is assumed to consume 10 percent of the ingot. The accident could occur as a result of a container drop during handling, a drop during inspection, or due to an earthquake. The release occurs at ground level. A handling accident resulting in the drop of a uranium ingot may be anticipated to occur over the life of the project (or about 1 in 10 years). The conditional probability of a fire that consumes 10 percent of the dropped ingot was assumed to be 1 in 10 drops at most. The estimated frequency of the accident is therefore 0.01 per year.

The material at risk is one 6-kilogram ingot of uranium. The damage ratio is 0.1, as it was assumed that 10 percent of the ingot would be consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1 for metal fires (DOE 1994b). The accident was assumed to occur outdoors or with little confinement. A leak path factor of 1 was assumed. This information is summarized in **Table F–58**.

Table F–58 Toxic Chemical Source Term for a Uranium Handling Accident

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Released (kilograms)*
Uranium	6	0.1	0.0001	1	1	0.00006

F.3.2.1.2 Design-Basis Uranium Fire

Uranium ingots (20 percent enrichment or less) from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to onsite storage at the Zero Power Physics Reactor Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms (310 pounds) of ingots. Depleted uranium also is stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a fire consuming the equivalent of one container of uranium (140 kilograms). The accident could occur due to a handling accident, poor housekeeping in the storage area, electrical failure, or an earthquake. The uranium is in the form of ingots that have a small surface-area-to-mass ratio. Uranium is stored in metal containers that are not combustible. The postulated accident was estimated to have a frequency of 1×10^{-5} per year (see the discussion of radiological accidents in Section F.2).

The material at risk is one 140-kilogram container of uranium. The damage ratio is 1, as it was assumed that all of the uranium would be consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1 for metal fires (DOE 1994b). The accident was assumed to occur outdoors or with little confinement (e.g., an open storage facility door). A leak path factor of 1 was assumed. This information is summarized in **Table F–59**.

Table F–59 Toxic Chemical Source Term for a Uranium Fire

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Uranium	140	1	0.0001	1	1	0.014

F.3.2.1.3 Design-Basis Earthquake – Multifacility Effects

This event is the same event as described under radiological accidents for the electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. The material at risk and release fraction values are summarized in **Table F-60**.

Table F–60 Toxic Chemical Source Term for a Design-Basis Earthquake

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Uranium	17	1	2.5×10^{-6}	2.5×10^{-6}	1	0.000043

F.3.2.1.4 Beyond-Design-Basis Earthquake – Multifacility Effects

This event is the same event as described under radiological accidents for electrometallurgical treatment at ANL-W. The airborne release fraction and respirable fraction values for cadmium are each 2.5×10^{-6} (Slaughterbeck et al. 1995). The material at risk and release fraction values are summarized in **Table F–61**.

Table F–61 Toxic Chemical Source Term for a Beyond-Design-Basis Earthquake

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Cadmium	1,000	1	2.5×10^{-6}	2.5×10^{-6}	1	0.0025
Uranium	17	1	2.5×10^{-6}	2.5×10^{-6}	1	0.000043

F.3.2.1.5 Liquid Sodium Fire

This event is the same event as described under radiological accidents for melt and dilute processing at ANL-W. The accident is associated with the fuel cleaning process used during the melt and dilute process or in preparation of the fuel for shipment to SRS for processing.

The accident involves a fire during the declad and clean processing of the spent nuclear fuel due to a breach of the Hot Fuel Examination Facility and exposure of liquid sodium to the air. The most probable cause of air in-leakage is expected to be an earthquake. As discussed in the radiological accident description, this event was assumed to occur with a frequency of 0.008 per year. The material at risk would be the sodium cleaned from the spent nuclear fuel and was conservatively estimated to be half of all of the sodium contained in the spent nuclear fuel, 300 kilograms. The release fraction values are provided in **Table F–62**. The assumption that all of the sodium would be converted to sodium hydroxide and volatilized by the fire results in the airborne release fraction and respirable fraction values of 1 each.

Table F–62 Toxic Chemical Source Term for a Sodium Fire in the Hot Fuel Examination Facility

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Sodium	330	1	1	1	0.125	41.3

F.3.2.2 Toxic Chemical Accidents at SRS

The SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) analyzed the consequences of accidental releases of hazardous chemicals for operations located in F-Area. These accidents involve the spill of materials associated with the wet storage of spent nuclear fuel in F-Area. These are generic-type accidents that are independent of processing cleaned and declad blanket fuel pins at either F-Canyon or Building 105-L. The activities associated with processing the cleaned and declad blanket spent nuclear fuel are not expected to result in the introduction of additional hazardous materials or additional accident scenarios. Therefore, the accident scenarios identified in the SRS Spent Nuclear Fuel Management Draft EIS were selected to represent the hazardous chemical accidents associated with processing sodium-bonded spent nuclear fuel.

F.3.3 Accident Analyses Consequences and Risk Results

Tables F–63 through **F–67** provide the chemical risk calculation results for electrometallurgical treatment process-related accidents at the ANL-W facility. **Table F–68** reproduces the consequences from hazardous chemical accidents at SRS, as originally developed for the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

Table F–63 Summary of Toxic Chemical Exposure Results for a Uranium Handling Accident at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Uranium	0.000177	0.000295	0.6 mg/m^3
Maximally exposed offsite individual	Uranium	1.14×10^{-8}	1.9×10^{-8}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–64 Summary of Toxic Chemical Exposure Results for a Uranium Fire at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Uranium	0.0413	0.0688	0.6 mg/m^3
Maximally exposed offsite individual	Uranium	2.7×10^{-6}	4.4×10^{-6}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–65 Summary of Toxic Chemical Exposure Results for a Design-Basis Earthquake at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Uranium	100 meters: 1.29×10^{-7} 230 meters: 1.03×10^{-6}	100 meters: 2.15×10^{-7} 230 meters: 1.72×10^{-6}	0.6mg/m^3
Maximally exposed offsite individual	Uranium	5.25×10^{-8}	8.75×10^{-8}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–66 Summary of Toxic Chemical Exposure Results for a Beyond-Design-Basis Earthquake at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Cadmium	7.5×10^{-6}	0.00025	0.03 mg/m^3
	Uranium	1.27×10^{-7}	2.12×10^{-7}	0.6 mg/m^3
Maximally exposed offsite individual	Cadmium	3.10×10^{-6}	0.0001	0.03 mg/m^3
	Uranium	5.3×10^{-8}	8.8×10^{-8}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–67 Summary of Toxic Chemical Exposure Results for a Sodium Fire at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of PEL-TWA*	*PEL-TWA*
Noninvolved worker	Sodium hydroxide	0.15	0.075	2 mg/m^3
Maximally exposed offsite individual	Sodium hydroxide	0.002	0.001	2 mg/m^3

PEL-TWA = Permissible Exposure Limits–Time-Weighted Average, mg/m^3 = milligrams per cubic meter.

[a] No ERPG value is available for sodium hydroxide; therefore, PEL-TWA was used instead.

Table F–68 Summary of Toxic Chemical Exposure Results for a Wet Storage Container Rupture at SRS

Frequency (event/year)	*Receptor*	*Chemical*	*Concentration* [a]	*Fraction of PEL-TWA*	*PEL-TWA*
0.005	Noninvolved worker	Sodium hydroxide	less than PEL-TWA	N/A[b]	2 mg/m^3
0.005	Noninvolved worker at 640 meters	Nitric acid	3.1×10^{-3} mg/m^3	0.00062	5 mg/m^3
	Maximally exposed offsite individual		4.0×10^{-4} mg/m^3	0.00008	5 mg/m^3
0.005	Noninvolved worker	Sodium nitrite	6.0×10^{-3} mg/m^3	0.0012 [2]	2 mg/m^3 [c]

PEL-TWA = Permissible Exposure Limits–Time-Weighted Average, mg/m^3 = milligrams per cubic meter.
[a] SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).
[b] Not available – SRS Spent Nuclear Fuel EIS states that concentration only in less than the lowest PEL-TWA.
[c] No PEL-TWA for this specific chemical. Lowest PEL-TWA of potential chemical reaction products is 2 milligrams per cubic meter.

Table F-69 provides a summary of the applicability of the analyzed toxic chemical accidents to each of the alternatives considered in detail for processing the sodium-bonded spent nuclear fuel. The hazardous chemical accidents applicable to the No Action Alternative include only those accidents associated with operation at ANL-W. Additionally, only three of the four accidents identified, excluding the beyond-design-basis earthquake, can be associated with this alternative. Accidents associated with this alternative are the result of activities from the final processing of the sodium-bonded spent nuclear fuel treated with the electrometallurgical treatment process as part of the Electrometallurgical Treatment Demonstration Program. Alternatives 2 through 5 include electrometallurgical treatment of at least some of the sodium-bonded spent nuclear fuel and decladding and cleaning of blanket spent nuclear fuel; therefore, all of the identified toxic chemical accidents at ANL-W are applicable to these alternatives. Alternative 1 includes electrometallurgical treatment of fuel, but no decladding and cleaning operations; therefore, for this alternative, all ANL-W accidents except the sodium fire are applicable. Processing of the spent nuclear fuel at SRS occurs only in Alternatives 3 and 5, and the accidents at SRS are applicable to these two alternatives. The accidents identified for ANL-W are associated with the electrometallurgical treatment of the sodium-bonded spent nuclear fuel. Alternative 6 does not include this treatment option and no other accidental releases of hazardous chemicals were identified.

Table F–69 Applicability of Hazardous (Toxic) Chemical Accidents to Sodium-Bonded Spent Nuclear Fuel Alternatives

	Alternative	*ANL-W Toxic Chemical Accidents*	*SRS Toxic Chemical Accidents*
	No action	Uranium handling accident Uranium fire Design-basis earthquake	Not applicable
1	Electrometallurgically treat blanket and driver fuel at ANL-W	Uranium handling accident Uranium fire Design-basis earthquake Beyond-design-basis earthquake	Not applicable
2	Clean and package blanket fuel in high-integrity cans and electrometallurgically treat driver fuel at ANL-W	Alternative 1 accidents plus sodium fire	Not applicable
3	Declad and clean blanket fuel and electrometallurgically treat driver fuel at ANL-W; PUREX process blanket fuel at SRS	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
4	Melt and dilute blanket fuel and electrometallurgically treat driver fuel at ANL-W	Alternative 1 accidents plus sodium fire	Not applicable
5	Declad and clean blanket fuel and electrometallurgically treat driver fuel at ANL-W; melt and dilute blanket fuel at SRS	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
6	Melt and dilute blanket and driver fuel at ANL-W	Sodium fire	Not applicable

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